Agreement (the Guaranteed Amount) is the Receivership Date Accumulated Book Value of the GIC, which is $1,034,447.59, less the sum of GIC Proceeds (cash proceeds actually received by the Plan from Confederation Life or any other entity making payment with respect to Confederation Life's obligations under the terms of the GIC, or from the sale or transfer of the GIC to unrelated third parties) and Advances under the Agreement as described below, plus interest on the net of the foregoing amount after the Receivership date at the Contract Rate of 7.15 percent.

The Advances: On the monthly occasions when the Employer, as Plan administrator, would otherwise request a withdrawal from the GIC to fund Withdrawal Events with respect to Account balances invested in the GIC, the Employer will instead notify the Trustee of the requested withdrawal amount. The Trustee will then determine whether it can satisfy the withdrawal request by using the assets in the G.I. Fund other than the GIC. If the Trustee determines that the funds available from the G.I. Fund are insufficient to honor the withdrawal request, the Trustee will determine the amount of additional funds necessary to honor the withdrawal request, and the Employer will make an Advance in that amount to the Plan. Valuation of the Account balances invested in the GIC for purposes of the Advances will be based on the Guaranteed Amount as described above.

Final Advance: The Agreement provides for a final Advance after the completion of the Receivership. After the Trustee has determined that the Plan will not receive any further proceeds from Confederation Life or its successors with respect to the GIC, the Employer shall make a final Advance to the Plan in the amount necessary to enable the Plan's recovery of the Guaranteed Amount. In the event the Receivership extends beyond the year 2000, the Employer will make the final Advance on the first business day in the year 2001 in the amount required on such date to enable the Plan to recover the Guaranteed Amount.

The Repayments: The Agreement provides that the Repayments of the Advances are restricted to the principal amounts of the Advances, and the Plan will pay no interest and will incur no expenses with respect to the Advances. The Repayments may be made only from the GIC Proceeds received by the Plan. No other Plan assets will be available for the Repayments. If the GIC Proceeds are not sufficient to repay fully the Advances, the Agreement provides that the Employer will have no recourse against the Plan, or against any participants or beneficiaries of the Plan, for the unpaid amount. To the extent the Plan receives GIC Proceeds in excess of the total amount of the Advances, such additional amounts will be retained by the Plan and allocated among the Accounts invested in the G.I. Fund.

6. In summary, the applicant represents that the proposed transaction satisfies the criteria of section 408(a) of the Act for the following reasons: (1) The Advances enable the Plan to resume the full funding of the Withdrawal Events; (2) The Advances will protect the Plan's investment in the GIC and will ensure that the Plan will recover all amounts due under the terms of the GIC; (3) The Plan will pay no interest or incur any expenses with respect to the Advances; (4) Repayment of the Advances will be made only from GIC Proceeds and no other Plan assets will be involved in the transactions; (5) Repayment of the Advances will be waived to the extent the Plan recoups less from the GIC Payors than the total amount of the Advances; and (6) In the event the Plan receives GIC Proceeds in excess of the Guaranteed Amount, such amounts will be retained by the Plan and allocated among the Accounts.

FOR FURTHER INFORMATION CONTACT: Ronald Willett of the Department (202) 219–8881. (This is not a toll-free number.)

General Information

The attention of interested persons is directed to the following:

(1) The fact that a transaction is the subject of an exemption under section 408(a) of the Act and/or section 4975(c)(2) of the Code does not relieve a fiduciary or other party in interest of disqualified person from certain other provisions of the Act and/or the Code, including any prohibited transaction provisions to which the exemption does not apply and the general fiduciary responsibility provisions of section 404 of the Act, which among other things require a fiduciary to discharge his duties respecting the plan solely in the interest of the participants and beneficiaries of the plan and in a prudent fashion in accordance with section 404(a)(1)(b) of the act; nor does it affect the requirement of section 401(a) of the Code that the plan must operate for the exclusive benefit of the employees of the employer maintaining the plan and their beneficiaries;

(2) Before an exemption may be granted under section 408(a) of the Act and/or section 4975(c)(2) of the Code, the Department must find that the exemption is administratively feasible, in the interests of the plan and of its participants and beneficiaries and protective of the rights of participants and beneficiaries of the plan;

(3) The proposed exemptions, if granted, will be supplemental to, and not in derogation of, any other provisions of the Act and/or the Code, including statutory or administrative exemptions and transitional rules. Furthermore, the fact that a transaction is subject to an administrative or statutory exemption is not dispositive of whether the transaction is in fact a prohibited transaction; and

(4) The proposed exemptions, if granted, will be subject to the express condition that the material facts and representations contained in each application are true and complete, and that each application accurately describes all material terms of the transaction which is the subject of the exemption.

Ivan Strasfel,

Director of Exemption Determinations, Pension and Welfare Benefits Administration, U.S. Department of Labor.

[FR Doc. 95–11536 Filed 5–9–95; 8:45 am]

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UNITED STATES NUCLEAR REGULATORY COMMISSION

Biweekly Notice

Applications and Amendments to Facility Operating LicensesInvolving No Significant Hazards Considerations

I. Background

Pursuant to Public Law 97-415, the U.S. Nuclear Regulatory Commission (the Commission or NRC staff) is publishing this regular biweekly notice. Public Law 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from April 17, 1995, through April 28, 1995. The last biweekly notice was published on April 26, 1995.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day notice period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish in the **Federal Register** a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

Written comments may be submitted by mail to the Rules Review and Directives Branch, Division of Freedom of Information and Publications Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this **Federal Register** notice. Written comments may also be delivered to Room 6D22, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland from 7:30 a.m. to 4:15 p.m. Federal workdays. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC. The filing of requests for a hearing and petitions for leave to intervene is discussed below.

By June 9, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment request involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public

Document Room, the Gelman Building, 2120 L Street, NW., Washington DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

Carolina Power & Light Company, et al., Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of amendment request: April 5, 1995

Description of amendment request: The licensee proposes to revise Technical Specification (TS) 3/4.9, Refueling Operations, to be consistent with NUREG-1431, Standard Technical Specifications, Westinghouse Plants, and to relocate the applicable sections from the TS that do not meet the Commission's screening criteria for retention.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

This change does not involve a significant hazards consideration for the following reasons:

The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes will have no significant impact on the safety, reliability, or operation of fuel handling equipment or activities. These changes will simplify the Technical Specifications and implement the recommendations of the Commission's Final Policy Statement on Technical Specification Improvements based upon the assumptions and analyses contained in the bases of NUREG-1431. Those elements that involve relocations to plant procedures are administrative in nature and do not involve any modifications to plant equipment or operation. Therefore, there would be no increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not introduce any new equipment or require existing equipment to operate to perform a function different from that previously evaluated in the Final Safety Analysis Report or Technical Specifications. The changes are consistent with the new Standard Techical Specification and assumptions contained in NUREG-1431 and in the Commission's Final Policy Statement on Technical Specification Improvements. Therefore, the proposed changes would not increase the possibility of a new or different type of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety.

The proposed changes do not affect any of the parameters which relate to the margin of safety as described in the [Bases] of the Technical Specifications or the Final Safety Analysis Report. Accordingly, NRC Acceptance Limits are not affected by these changes. For those specifications being relocated to other plant documents, these changes are purely administrative. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, Post Office Box 1551, Raleigh, North Carolina 27602

NRC Project Director: David B. Matthews

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, Illinois, Docket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment request: September 15, 1992, as supplemented April 21, 1995

Description of amendment request: As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom Technical Specifications (TSs).

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TSs for both Dresden and Quad Cities against the Standard Technical Specifications (STSs) contained in NUREG-0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing TSs to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TSs to the STSs contained in NUREG-0123.

The TSUP for Dresden and Quad Cities is not a complete adaption of the STSs. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TSs based on the licensee's responses to Generic Letters (GLs), and (4) relocating specific items to more appropriate TS locations.

The application dated September 15, 1992, as supplemented April 21, 1995, proposed to upgrade only Sections 2.0 (Safety Limits and Limiting Safety System Settings), 3/4.11 (Power Distribution Limits), and 3/4.12 (Special Test Exceptions) of the Dresden and Quad Cities TS.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Section 2.0

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed changes to Specifications 1/2.1 and 1/2.2 to delete the present Applicability and Objective sections represent administrative changes to format and presentation of material. The proposed changes provide the user with a format that will allow better access to needed information and provides concise Safety Limit, Limiting Safety System Settings, Applicability and Action requirements. The additions of Applicability and Action requirements represent clarification of intended requirements that do not presently state all required conditions of operability or provide clearly stated Action statements if the requirements are not met. The combining of the two sections and added requirements follow STS guidelines that are in use at many operating BWRs with similar design and operating configurations as Dresden and Quad Cities Stations. Operability requirements for Safety Limits have been chosen to reflect only those Operational Modes where the Safety Limits apply. Operability requirements for Limiting Safety System Settings are already stated in other sections of the Technical Specifications, thus reference to the appropriate operability requirement is made rather than repeating the requirement in the Limiting Safety System Setting Specification.

Deletion of the Power Transient Safety Limit does not impact any safety analyses. The safety analyses assume the Reactor Protection System (RPS) operates as designed and the reactor scrams when the neutron flux exceeds the limiting safety system setting. The proposed Technical Specifications will continue to provide a highly reliable system to operate as assumed in the safety analyses. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The reactor water level low scram setpoint is changed (for Quad Cities) to be consistent with other reactor water level setpoints in the Technical Specifications and the STS. The setpoint is equivalent to the current requirement but is expressed as the reactor water level above the top of active fuel.

The scram discharge volume scram level is converted for Dresden Unit 2 and Unit 3 to gallons to be consistent with the Quad Cities Units. The proposed setpoints are consistent with the current specifications. The change in the units does not represent a change in the physical setpoint.

The proposed change to delete the APRM Downnscale Scram trip function for Quad Cities has been evaluated by Commonwealth Edison and General Electric and previously approved for Dresden Station. The events of concern with respect to the APRM/IRM companion trip are the Control Rod Drop Accident and the low power Rod Withdrawal Error. The FSAR and reload safety analyses do not credit this scram function in the termination of either of these events. Since this scram function is not credited in the termination of these events, the elimination of this scram function has no adverse effect on previously evaluated accidents.

The change to the low condenser vacuum scram setpoint from 23 inches Hg to 21 inches of Hg is consistent with an identical change made to Quad Cities Units 1 and 2. The low condenser vacuum scram is an anticipatory scram and is not credited in any transient analysis. Thus the reduction in the setpoint will not affect any transient analysis.

The proposed changes do not alter the intent of existing setpoints or accident assumptions and follow existing requirements at other operating BWRs for operability and Action statements. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated because:

The proposed administrative changes to the format and arrangement of material do not affect technical requirements or assumptions of any potential accident and; therefore, cannot create the possibility of a new or different kind of accident from any previously evaluated.

The proposed addition of Applicability and Action requirements enhance the understanding and usability of the Technical Specifications and thus represent an improvement over present specifications. New requirements are modeled after those in use at operating BWRs and do not represent requirements that will adversely affect potential accident analyses or assumptions. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Deletion of the Power Transient Safety Limit does not involve a change in the design or operation of any systems assumed to operate in the safety analyses. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The change in the units for the Reactor Water Level scram function do not change any physical plant setpoints. The setpoint will remain the same but will be expressed as the level above the top of active fuel. The change does not create the possibility of a new or different kind of accident.

The conversion of the Scram Discharge Volume scram setpoint from inches to gallons does not alter any physical plant setpoints. The setpoint will remain the same but will be expressed in gallons rather than inches. The change will provide consistency between Dresden and Quad Cities.

The deletion of the APRM Downscale Scram Trip Function does not introduce any new accident. The limiting accidents, Control Rod Drop, Rod Withdrawal Error, in the operating region of transition between the Startup and Run Operational Modes are well understood and are evaluated in FSAR and reload analyses. Other control rod initiated events which are less limiting in this region are subsets of the low power Rod Withdrawal Error event and are bounded by it and the design basis Control Rod Drop Accident. General Electric has indicated that, for reactivity insertion mechanisms at very low power, the only effect of the deletion of the APRM downscale scram would be that the initial power level could be a few percent lower which would not have a significant effect on the severity of the event. In addition, proper overlap between the IRMs and APRMs is not affected since the calibration requirements are not being changed.

The change in the low condenser vacuum scram function will not create the possibility of a new or different kind of accident because the function is not recognized in any of the transient analysis. The low condenser vacuum scram function is an anticipatory scram.

The proposed changes do not involve a significant reduction in the margin of safety because:

The proposed administrative changes to format, arrangement of material, clarification of requirements and other non-technical changes do not affect any safety aspects of the plant and as such can not involve a significant reduction in the margin of safety.

The proposed Applicability statements require availability of Safety Limits and Limiting Safety System Settings when required to perform their respective functions. Proposed Actions for Safety Limits allow only 2 hours to be in Hot Shutdown and then reference Specification 6.4 to ensure that proper reports are made and restart is prohibited until approved by the NRC. These provisions help ensure that present margins are not significantly reduced.

Deletion of the Power Transient Safety Limit does not impact the margin assumed in the safety analyses. The safety analyses assume the RPS operates as designed and the reactor scrams when the neutron flux exceeds the limiting safety system setting. The margins assumed in the design of the RPS and in the safety and transient analyses calculations have not been revised. Therefore, this change does not involve a significant reduction in the margin of safety.

The change in units to the Reactor Water Level scram setpoint and the Scram Discharge Volume scram setpoint do not involve a significant reduction in the margin of safety because the changes do not represent a change in the physical setpoints.

The reduction in the Low Condenser Vacuum scram setpoint does not represent a reduction in the margin of safety because the scram is not credited in any transient analysis.

The APRM Downscale Scram Trip Function is not credited in the termination of any FSAR or reload safety analysis event. As such, the elimination of this scram function has no effect on any margin of safety.

Section 3/4.11

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These proposed changes are consistent with the current safety analyses and have been previously determined to represent sufficient requirements for the assurance of reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The Generic Changes to the technical specifications involve administrative changes to format and arrangement of the material. As such, these changes cannot involve a significant increase in the probability or consequences of an accident previously evaluated.

The current specifications require the reactor to be placed in cold shutdown when a thermal limit was exceeded and not restored within the allotted 2 hours, but the proposed specifications require the reactor to be less than 25% of rated thermal power if this condition occurred. The change eliminates a shutdown and requires the power level to be reduced to the point that the limits are no longer applicable.

Therefore, the change will not increase the probability or consequences of an accident.

Create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the stations; however, these changes provide additional restrictions which are in accordance with the current safety analyses, or are to provide for additional testing or surveillance which will not introduce new failure mechanisms beyond those already considered in the current safety analyses. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

Since the Generic Changes proposed to the technical specifications are administrative in nature, they cannot create the possibility of a new or different kind of accident from any previously evaluated.

The requirement to reduce thermal power to less than 25% of rated thermal power rather than place the reactor in cold shutdown will not create a new or different kind of accident because the thermal limits are not required in operational mode 1 when thermal power is less than 25% of rated power.

Involve a significant reduction in the margin of safety because:

In general, the proposed changes represent the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the latter individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain within their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The Generic Changes proposed in this amendment request are administrative in nature and, as such, do not involve a reduction in the margin of safety.

Section 3/4.12

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed Specification 3/4.12 is a new section which will provide the user with a format that will allow better access to needed information and provide concise Applicability and Action requirements. The additions of Applicability and Action requirements represent classification of intended requirements that do not presently state all required conditions of operability or provide clearly stated Action statements if the requirements are not met. The combining of the two sections and the added requirements follow Standard Technical Specifications (STS) guidelines that are in use at many operating BWRs with similar design and operating configurations as Dresden and Quad Cities Stations.

The proposed Section 3/4.12 involves the relocation of present requirements into one section identical to STS provisions. The changes also implement the Applicability and Action provisions of the STS and later operating BWR plants that have been evaluated and found acceptable for use at Dresden and Quad Cities. Present Surveillance Requirements are replaced, where applicable, with proven STS guidelines that are being used at plants with a system similar to that at Dresden and Quad Cities. The changes in the present Surveillance Requirements add testing requirements that are not presently in the Dresden and Quad Cities technical specifications. The proposed changes do not affect accident assumptions other than a minor increase in the initial power level (approximately 0.2% to 1%) and as such, do not involve a significant increase in the probability of an accident previously evaluated. The proposed specifications add additional requirements to specifications currently contained in the Technical Specifications. Since the proposed changes to the Technical Specifications implement requirements that have been demonstrated to provide acceptable operability provisions at other facilities with a design similar to that at Dresden and Quad Cities, the proposed changes do not significantly increase the consequences of an accident previously evaluated.

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated because:

The proposed administrative changes to the format and arrangement of material do not affect technical requirements or assumptions of any potential accident and; therefore, cannot create the possibility of a new or different kind of accident from any previously evaluated.

The proposed addition of Applicability and Action requirements enhance the understanding and usability of the Technical Specifications and thus represent an improvement over present specifications. New requirements are modeled after those in use at operating BWRs and do not represent requirements that will adversely affect potential accident analyses or assumptions. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not involve a significant reduction in the margin of safety because:

The proposed administrative changes to format, arrangement of material, clarification of requirements and other non technical changes do not affect any safety aspects of the plant and as such can not involve a significant reduction in the margin of safety.

In addition, the commission has provided guidance concerning the application of standards for determining whether significant hazards consideration exists by providing certain examples (51 FR 7751) of amendments that are considered not likely to involve significant hazards considerations. Commonwealth Edison has reviewed the proposed changes against these examples and believes that the proposed changes fall within the scope of example (ii) ‘‘a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications’’.

The proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

The NRC staff has reviewed the licensee’s analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: for Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603

NRC Project Director: Robert A. Capra

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Units 2 and 3, Grundy County, IllinoisDocket Nos. 50-254 and 50-265, Quad Cities Nuclear Power Station, Units 1 and 2, Rock Island County, Illinois

Date of application for amendment request: December 15, 1993, as supplemented by letter dated April 21, 1995

Description of amendment request: As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom Technical Specifications (TSs) used.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TSs for both Dresden and Quad Cities against the Standard Technical Specifications (STSs) contained in NUREG-0123, ''Standard Technical Specifications General Electric Plants BWR/4.'' The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing TSs to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TSs to the STSs contained in NUREG-0123.

The TSUP for Dresden and Quad Cities is not a complete adaption of the STSs. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TSs based on the licensee's responses to Generic Letters (GLs), and (4) relocating specific items to more appropriate TS locations.

The December 15, 1993, and April 21, 1995, applications proposed to upgrade only Section 5.0 (Design Features) of the Dresden and Quad Cities TSs.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide continued assurance that specified [parameters remain] within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident. Some of the proposed changes to the current Technical Specifications (CTS) represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for current Dresden and Quad Cities Station's Technical Specifications Section 5.0 represent a minor relaxation of the current requirements, and is based on BWR-STS (NUREG-0123) guidelines or later operating BWR plant's NRC accepted changes. The proposed changes are consistent with the current safety analyses and have been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis. Any deviations from CTS or STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations.

Details describing the plant's design are presented in TSUP Section 5.0. There are no Limiting Conditions for Operation (LCO) or Surveillance Requirements (SR) encompassed within TSUP Section 5.0. This information is administrative in nature and consistent to the UFSAR; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment.

Create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor relaxations of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. The proposed changes are administrative in nature and do not involve a revision in the operation of the station. As such, there are no changes to the current safety analysis. Therefore, the proposed changes will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment for Dresden and Quad Cities Station's Technical Specifications Section 5.0 is based on BWR-STS guidelines or later operating BWR plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden or Quad Cities Nuclear Power Stations considering similarity of system or component design versus the BWR-STS or later operating BWRs. Any deviations from CTS or BWR-STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden and Quad Cities Stations. No new modes of operation are introduced by the proposed changes. The proposed changes maintain at least the present level of operability, and in some cases are more conservative. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Involve a significant reduction in the margin of safety because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Others represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment to Technical Specification Section 5.0 implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from CTS or BWR-STS requirements do not significantly reduce the margin of safety for Dresden or Quad Cities Stations. These changes do not involve revisions to the design of the station. The proposed changes are administrative in nature and do not involve a revision in the operation of the station. As such, there are no changes to the current safety analysis. Therefore, the proposed changes will not introduce new failure mechanisms beyond those already considered in the current safety analyses. Therefore, because the proposed changes are administrative in nature, do not involve a revision in the operation of the station and maintains the current design requirements specified in the UFSAR, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: For Dresden, Morris Area Public Library District, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603

NRC Project Director: Robert A. Capra

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of amendment request: December 13, 1994

Description of amendment request: The proposed amendment would revise the Palisades' technical specifications (TSs) to add a high thermal performance (HTP) departure from nucleate boiling correlation to Safety Limit 2.1. The HTP correlation is used for the high thermal performance fuel loaded during recent fuel cycles.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the TS adds the HTP critical heat flux correlation to the Safety Limit - Reactor Core Section 2.1. The HTP correlation is an NRC approved methodology for a Departure from Nucleate Boiling (DNB) Correlation for high thermal performance (HTP) fuel as is used at Palisades. The HTP correlation is an extension of the currently approved ANFP correlation. There are no associated changes in plant operation. Palisades fuel loaded in cycle 9 and later meet the requirements of the HTP correlation. Therefore, operation of the facility in accordance with the proposed TS would not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any previously evaluated.

The HTP correlation will allow for more accurate DNB predictions within the applicable operating conditions for fuels with the HTP design used at Palisades. There are no changes in plant operation. Therefore operation of the facility in accordance with the proposed TS would not create the possibility of a new or different kind of accident from any previously evaluated.

3. Involve a significant reduction in a margin of safety.

As stated previously, the HTP correlation will allow for more accurate DNB predictions within the applicable operating conditions for fuel with the HTP design. There are no associated changes in plant operation. Therefore, operation of the facility in accordance with the proposed TS would not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Attorney for licensee: Judd L. Bacon, Esquire, Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201

NRC Project Director: Cynthia A. Carpenter, Acting

Duke Power Company, et al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: January 18, 1995

Description of amendment request: The proposed amendments would relocate the requirements for the seismic instrumentation, meteorological instrumentation, and loose-part detection system from the Technical Specifications to the Selected Licensee Commitment (SCL) Manual. This will allow future changes to these controls to be performed under the provisions of 10 CFR 50.59. No changes are being made to the technical content of the affected Technical Specification pages.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1

The requested amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated. Relocation of the affected TS sections to the SLC Manual will have no effect on the probability of any accident occurring. In addition, the consequences of an accident will not be impacted since the above instrumentation will continue to be utilized in the same manner as before. No impact on the plant response to accidents will be created.

Criterion 2

The requested amendments will not create the possibility of a new or different kind of accident from any accident previously evaluated. No new accident causal mechanisms will be created as a result of relocating the affected TS requirements to the SLC Manual. Plant operation will not be affected by the proposed amendments and no new failure modes will be created.

Criterion 3

The requested amendments will not involve a significant reduction in a margin of safety. No impact upon any plant safety margins will be created. Relocation of the affected TS requirements to the SLC Manual is consistent with the content of the Westinghouse RSTS [Revised Standard Technical Specifications], as the NRC did not require technical specification controls for the affected instrumentation in the RSTS. The proposed amendments are consistent with the NRC philosophy of encouraging utilities to propose amendments that are consistent with the content of the RSTS.

Based upon the preceding analyses, Duke Power Company concludes that the requested amendments do not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242

NRC Project Director: Herbert N. Berkow

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of amendment request: April 3, 1995

Description of amendment request: The amendments will incorporate line-item TS improvements to Specifications 3/4.8.1 ''Electrical Power Systems-A.C. Sources,'' and 4.8.1.2.2 ''Electrical Power Systems-Shutdown.'' The proposed changes are consistent with recommendations for Emergency Diesel Generator (EDG) Surveillance Requirements in NUREG-1366, and regulatory guidance provided in Generic Letter (GL) 93-05 and GL 94-01. This proposal also contains FPL's commitment to implement a maintenance program for monitoring and maintaining EDG performance for both St. Lucie Units consistent with 10 CFR 50.65 and the guidance of Regulatory Guide 1.160.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

(1) Operation of the facility in accordance with the proposed amendment would not

involve a significant increase in the probability or consequences of an accident previously evaluated.

The license amendments proposed for St. Lucie Units 1 and 2 will incorporate line-item Technical Specification (TS) improvements for Emergency Diesel Generators (EDG) pursuant to guidance provided in Generic Letters (GL) 93-05 and 94-01. The EDGs are not accident initiators, the proposed TS changes do not involve any assumptions relative to accident initiators in the plant safety analyses, and therefore the proposed amendments will not impact the probability of occurrence for accidents previously analyzed.

The EDG line-item TS improvements associated with GL 93-05 are based on recommendations designed to remove unwarranted requirements for testing during power operation and other factors that are counter-productive to safety in terms of equipment degradation and availability. These recommendations resulted from a comprehensive study of industry-wide EDG surveillance requirements and subsequent findings reported by the NRC in NUREG-1366. The proposed amendments are consistent with the GL 93-05 guidance for implementing such recommendations.

Similarly, GL 94-01 provides guidance for a line-item TS improvement that will remove accelerated testing requirements from the TS provided that the licensee commits to a maintenance program for monitoring and maintaining EDG performance that includes the applicable provisions of the maintenance rule (10 CFR 50.65). Such a program will further assure EDG availability. Since the availability of EDGs is assumed in certain success paths for mitigating analyzed accidents, an improvement in EDG availability will enhance accident mitigation capabilities.

Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendments incorporate line-item TS improvements to EDG surveillance testing requirements, and will not change the physical plant or the modes of plant operation defined in the Facility License. The changes do not involve the addition or modification of equipment, nor do they alter the design or methods of operation of plant systems. Plant configurations that are prohibited by TS will not be created by the amendments. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendments are designed to improve EDG availability by eliminating unwarranted surveillance testing. The presently specified surveillance intervals are not changed. The proposed changes do not otherwise alter the basis for any technical specification that is related to the establishment of, or the maintenance of a nuclear safety margin. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the above discussion and the supporting Evaluation of Technical Specification changes, FPL has determined that the proposed license amendment involves no significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003

Attorney for licensee: J. R. Newman, Esquire, Morgan, Lewis & Bockius, 1800 M Street, NW., Washington, DC 20036

NRC Project Director: David B. Matthews, Director

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: March 7, 1995

Description of amendment request: The proposed amendment would add an Exception to Technical Specifications (TS) 3.6.A and 3.6.C. The Exception would permit reduced component cooling water flow for short periods of time, while component cooling water heat exchangers are shifted.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The staff's review is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

Plant experience shows that the component cooling water heat exchangers can be shifted in a few minutes; well within the time limit for Remedial Action under this TS 3.6.A or C, or TS 3.0.A. Thus, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed change does not affect equipment reliability when such equipment is required to be operable. Existing TS 3.6 and its Remedial Action statement govern the plant circumstances under which cooling water subsystems are required, and specify the maximum time such subsystems may be unavailable. The proposed change does affects neither operating requirements nor the time limit on restoring system operability.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed change does not significantly alter the availability or condition of the cooling water subsystems and, therefore, does not alter the accident analysis or its associated conclusions. The proposed change would permit flow in one component cooling water train to be reduced below that required for operation of the emergency core cooling systems in the recirculation mode, for a short period of time. The amount of time that flow is reduced is small, and full flow operation can be easily restored within the time required for design heat load removal. Thus, there is no significant reduction in a margin of safety.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that this amendment request involves no significant hazards consideration.

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578

Attorney for licensee: Mary Ann Lynch, Esquire, Maine Yankee Atomic Power Company, 329 Bath Road, Brunswick, ME 04011

NRC Project Director: Phillip F. McKee

Northeast Nuclear Energy Company (NNECO), Docket No. 50-245, Millstone Nuclear Power Station, Unit 1, New London County, Connecticut

Date of amendment request: April 18, 1995

Description of amendment request: The proposed amendment would allow the use of the ANSI/ANS 5.1-1979 decay heat model for post-loss of coolant accident containment cooling analysis.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

NNECO has reviewed the proposed change in accordance with 10CFR50.92 and concluded that the change does not involve a significant hazards consideration (SHC). The basis for this conclusion is that the three criteria of 10CFR50.92(c) are not compromised. The proposed change does not involve an SHC because the change would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

The change to the decay heat model used to determine post-accident conditions cannot affect the probability of any accident. No changes to plant operation or design would occur due to the new analysis.

The new model cannot directly affect the consequences of an accident, since it is the tool used to predict the temperature effects of the postulated accident. However, using the ANSI/ANS 5.1-1979 model could change the anticipated actions necessary to respond to an event. Changing the response action could possibly affect the consequences of an accident. This model change will not have such an effect. Operator actions to throttle LPCI [low pressure coolant injection], CS [core spray], or ESW [emergency service water] pump flow are taken based upon observed conditions, not predetermined data points from the analysis.

Operability of the emergency core cooling systems (ECCS) can be shown for temperatures that are higher than those predicted by the containment cooling analysis.

Therefore, the utilization of the ANSI/ANS 5.1-1979 decay heat model does not involve a significant increase in the probability or consequences of a previously evaluated accident.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed license amendment only revises the predicted temperature that result from a postulated accident. There is no change to the design or operation of any system or component. Since this change only deals with the post-accident effects of currently analyzed accidents, there is no possibility of creating a new or different kind of accident.

3. Involve a significant reduction in the margin of safety.

The early design documentation stated that the ECCS components were designed for post-accident torus temperatures of 203°F. As this issue evolved, NNECO performed operability determinations which showed that peak temperatures of 209°F were acceptable. Utilizing a more accurate decay heat model which results in lower predicted peak temperatures demonstrates the acceptability of the plant design. Therefore, replacing the May-Witt decay heat model with the ANSI/ANS 5.1-1979 model does not result in a decrease in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of amendment request: March 29, 1995

Description of amendment request: The proposed amendment changes Technical Specifications to revise peaking factor penalties based on NRC approved methods.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed changes do not involve an SHC because the changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously analyzed.

The proposed changes to the action statements of Sections 3.2.2.1 and 3.2.2.2 are purely administrative and therefore they do not adversely affect the probability or consequences of an accident previously analyzed. The proposed changes to Surveillance Requirements 4.2.2.1.2.e, 4.2.2.1.4.e, 4.2.2.2.2.e and 4.2.2.2.4.e and Section 6.9.1.6.b are based on the NRC approved methodology for calculating the penalty to be applied to $F_Q^M(Z)$. The margin for the F_Q^{RTP} limit is still maintained by the proposed changes. In addition, the penalty is included in the COLR [Core Operating Limits Report] which will be maintained and controlled per the requirements of 10CFR50.59. Therefore, the proposed changes do not increase the probability or consequences of an accident previously analyzed.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The proposed changes to the Action Statement of Sections 3.2.2.1 and 3.2.2.2 are purely administrative and therefore, they do not create the possibility of a new or different kind of accident from any previously analyzed. The proposed changes to Surveillance Requirements 4.2.2.1.2.e, 4.2.2.1.4.e, 4.2.2.2.2.e, and 4.2.2.2.4.e and Section 6.9.1.6.b do not create a malfunction that is different from those previously evaluated. The changes do not involve positioning reactivity systems or plant components into any new configuration or sequence not previously analyzed. Therefore, the changes will not create the possibility of a new or different kind of accident from any other previously analyzed.

3. Involve a significant reduction in the margin of safety.

The proposed changes to the action statements of Sections 3.2.2.1 and 3.2.2.2 are purely administrative and therefore they will not reduce the margin of safety. The proposed changes to Surveillance Requirements 4.2.2.1.2.e, 4.2.2.1.4.e, 4.2.2.2.2.e and 4.2.2.2.4.e and Section 6.9.1.6.b do not reduce the margin to the F_Q^{RTP} limit. The approved methods more distinctly evaluate the expected changes to F_Q^M than previously existed. Therefore, there is no impact on the margin of safety as specified in the Technical Specifications.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resource Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New London Turnpike, Norwich, CT 06360.

Attorney for licensee: Ms. L. M. Cuoco, Senior Nuclear Counsel, Northeast Utilities Service Company, Post Office Box 270, Hartford, CT 06141-0270.

NRC Project Director: Phillip F. McKee

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-277, Peach Bottom Atomic Power Station, Unit No. 2, York County, Pennsylvania

Date of application for amendment: March 30, 1995

Description of amendment request: The proposed change would revise Technical Specifications Section 4.7.D.1.b.(1) by adding a footnote to exempt the High Pressure Coolant Injection [HPCI] motor-operated valve MO-2-23-015 from quarterly stoke testing requirements until refueling outage 2RO11.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or

consequences of an accident previously evaluated.

The proposed change does not serve as an initiator or contributor to any accidents previously evaluated. It does not decrease the effectiveness of equipment relied upon to mitigate previously evaluated accidents. A calculation was performed and it has been determined the leakage through the valve's packing will be within the allowable limits of containment leakage (L_a). While positioning the valve in the backseated position does increase its stroke time, it has been calculated and demonstrated that the valve will close within the TS time limit of 20 seconds.

Therefore, the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not serve as an initiator or contributor to any of the accidents previously evaluated. The proposed change does not introduce any new modes of plant operation.

Implementation of the proposed changes will not affect the design function or configuration of any component or introduce any new operating scenarios or failure modes or accident initiation. It does not impair or prevent safety systems from performing their safety function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety.

The proposed change does not serve as an initiator or contributor to any accidents evaluated in the [Safety Analysis Report] SAR. It has no impact on any safety analysis assumptions. Exempting the HPCI valve MO-2-23-015 from quarterly stroke testing until 2RO11 does not impact its reliability or affect its ability to perform its intended safety function. The change does not adversely affect the assumptions or sequence of events used in any accident analysis.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: March 16, 1995

Description of amendment request: This amendment would change the existing requirements for the Source Range Monitors (SRM) while the plant is in the refueling condition to requirements based on the Improved Technical Specifications in NUREG-1433, ''Standard Technical Specification General Electric Plants, BWR/4.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

The proposed changes to the SRM requirements will not increase the probability or consequences of an accident previously evaluated. The SRMs are not assumed to function during any UFSAR [Updated Final Safety Analysis Report] design basis accident or transient analysis. This TS change will not alter any safety limits which ensure the integrity of fuel barriers, and will not result in any increase to onsite or offsite dose. Additionally, continued availability of the SRMs in the refuel mode is ensured through additional testing requirements being added by this TS change. The changes to the SRM requirements will not alter the operation of equipment assumed to be available for the mitigation of accidents or transients.

The proposed changes are based on NUREG-1433, ''Standard Technical Specifications General Electric Plants, BWR/4,'' and are consistent with the PECO Energy submittal of September 29, 1994, requesting an overall conversion, based on NUREG-1433. The overall conversion to the ITS [Improved Technical Specifications] included both technically justified deviations from the NUREG, and technically justified changes from the PBAPS current TS.

2. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes to the SRM requirements will not create the possibility of a new or different type of accident from any previously evaluated. The SRMs are not assumed to function during any analyzed UFSAR design basis accident or transient analysis. Additionally, the changes will not involve any changes to plant systems, structures or components (SCCs) which could act as new accident initiators. Implementation of the proposed changes will effect the manner in which these SCCs are tested; however, TS requirements that govern routine testing and verification of plant components and variables are not assumed to be initiators of any analyzed event.

3. The proposed change does not result in a significant reduction in the margin of safety.

No margins of safety are reduced as a result of the proposed TS changes. No safety limits will be changed as a result of this TS change. The proposed change does not involve a reduction in the margin of safety because SRMs are not credited in any safety analysis. At least one SRM will remain operable during rod withdrawal during core alterations and rod withdrawal will not occur if no SRMs are operable. Excessive reactivity additions will be quickly identified and mitigated by the Intermediate Range Monitors and associated rod blocks. The Average Power Range Monitor Flux scram, and not any SRM function, is credited for mitigating a rod withdrawal or reactivity addition accident.

Use of a spiral offload or reload pattern will provide assurance that the SRM will be in the optimum position for monitoring changes in neutron flux levels during core alternations.

The changes proposed in this TS change do not introduce any hardware changes, and will not alter the intended operation of plant structures, systems or components utilized in the mitigation of accidents or transients. Additionally, these changes will not introduce any new failure modes of plant equipment not previously evaluated.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

PECO Energy Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Dockets Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: March 22, 1995

Description of amendment request: The amendment would revise Note (1) for Technical Specifications Tables 3.7.2 through 3.7.4 by reducing the Local Leak Rate Test (LLRT) hold time duration from one hour to 20 minutes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change does not serve as an initiator or contributor to any accidents previously evaluated. It does not decrease the effectiveness of equipment relied upon to mitigate previously evaluated accidents. The change does not involve any physical changes to any plant systems, structures, or components.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed changed does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed change does not serve as an initiator or contributor to any of the accidents previously evaluated. The proposed change does not introduce any new modes of plant operation.

Implementation of the proposed changes will not affect the design function or configuration of any component or introduce any new operating scenarios or failure modes or accident initiation. It does not impair or prevent safety systems from performing their safety function.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed changes do not involve a significant reduction in a margin of safety.

The proposed change does not serve as an initiator or contributor to any accidents evaluated in the SAR [Safety Analysis Report]. It has no impact on any safety analysis assumptions. Changing the LLRT duration hold time from one hour to 20 minutes does not impact equipment reliability. The change does not adversely affect the assumptions or sequence of events used in any accident analysis. Therefore, the propose change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, PECO Energy Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

Philadelphia Electric Company, Docket Nos. 50-352 and 50-353, Limerick Generating Station, Units 1 and 2, Montgomery County, Pennsylvania

Date of amendment request: November 21, 1994, as supplemented by letter dated April 6, 1995

Description of amendment request: The proposed amendment would make changes affecting the Administrative Controls Section of the Technical Specifications (TSs). The areas proposed to be changed are: 1) NEEDS [Nuclear Effectiveness and efficiency Design Study] Organization Title Changes, 2) Minimum Shift Crew Composition, 3) Delete Independent Techincal Review Section from TS, 4) Delete NRB [Nuclear Review Board] Review Section from TS, and 5) Delete NRB Audit Section from TS.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed Technical Specifications changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes to revise the organization position titles, PORC [Plant Operations Review Committee] composition description, and eliminate the Assistant Superintendent - Operations position do not involve any physical modifications to plant structures, systems, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. The proposed changes to position titles will not change the requirements for the qualifications and training of personnel in any management or supervisory position. Personnel will continue to meet the guidance specified in ANSI/ANS 3.1-1978 as required by Technical Specification 6.3.1. The probability of occurrence of an accident is based in part on: the training and qualifications of the personnel filling key plant management and supervisory positions; clear lines of authority, responsibility and communication; and, adequate management and corporate oversight of plant performance and activities. The proposed TS changes do not change any of these management and organizational elements.

Allowing the Plant Manager to designate appropriately qualified, trained and experienced members of the LGS [Limerick Generating Station] staff as members of the PORC, as proposed, will not degrade the effectiveness of the PORC. The qualifications, training and experience level of the PORC will meet the requirements listed in ANSI/ANS 3.1-1978, and the required PORC quorum (including the use of alternates) will not be affected.

Elimination of the position of Assistant Superintendent - Operations eliminates a level of supervision between the Plant Manager and the Shift Managers. The Shift Managers, who hold SRO licenses, will report directly to the Senior Manager - Operations. Other organizational changes within the Operations group (i.e., establishment of the positions of Manager - Operations Services and Manager - Operations Support) will ensure that the Senior Manager - Operations has sufficient time to properly supervise and monitor on-shift performance. The Senior Manager -Operations and/or an Operations Manager will be required to hold a Senior Reactor Operator (SRO) license. Individuals filling these positions will satisfy the applicable training, qualifications, and experience requirements of ANSI/ANS 3.1-1978.

The consequences of an accident could be affected by the qualifications and training of plant management and supervisory personnel. However, the proposed changes do not change the qualifications and training of personnel in any management or supervisory position. Personnel will continue to meet the criteria specified in ANSI/ANS 3.1-1978 as required by TS 6.3.1.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS changes to increase the minimum shift crew composition do not involve any physical changes to plant SSC.

The probability of the occurrence of an accident is based in part on the operating crew and their ability to safely operate the plant. The increase in the minimum on-shift crew composition and the associated changes improves the capability of the on-shift crew to safely operate the plant and SSC, thereby reducing the probability of a situation that could result in an accident. The increase in the minimum on-shift crew composition will improve the manner in which the SSC are operated, maintained, tested, and inspected.

The consequences of an accident could be affected by an operating error. However, the proposed TS changes increase the number of licensed operators required to be on-shift, and therefore, increase the capability of the on-shift crew to properly operate the facility and to implement the appropriate emergency procedures to reduce the consequences of an accident.

The proposed changes will also delete redundant and/or relocate existing independent technical review and, Nuclear Review Board review and audit requirements from TS that are and/or will be contained in the LGS UFSAR [Updated Final Safety Analysis Report]. Removal of redundant/relocation of existing requirements does not affect any equipment important to safety, or involve any physical modifications to plant SSC, therefore, is not associated with an accident initiator or accident mitigator and can not affect the probability of occurrence of an accident or increase the consequences of an accident. The licensee controlled UFSAR containing the requirements will be maintained using the provisions of 10 CFR 50.59, or 10 CFR 50.54(a), as appropriate, and are subject to the change control process in the Administrative Controls Section (6.0) of the Technical Specifications. Since future changes to related licensee-controlled documents will be evaluated per 10 CFR 50.59 or 10 CFR 50.54(a), no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed.

Therefore, these proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS changes to revise the organization position titles, PORC composition description, and eliminate the Assistant Superintendent - Operations position do not involve any physical modifications to plant structures, systems, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. The proposed changes to position titles will not change the requirements for the qualifications and training of personnel in any management or supervisory position. Personnel will continue to meet the guidance specified in ANSI/ANS 3.1-1978 as required by Technical Specification 6.3.1. Therefore, these proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the on-shift crew composition can not create the possibility of a new or different type of accident than previously evaluated in the SAR since implementation of the changes will not involve any physical changes to the plant SSC. The increase in the minimum on-shift crew composition increases the ability of the operating crew to ensure that the SSC are properly operated, maintained, tested and inspected. An increase in the required number of licensed operators on each shift improves the ability of the crew to adequately operate the facility, to respond to accident conditions, and to implement applicable plant procedures. Therefore, these proposed TS changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will also delete redundant and/or relocate existing independent technical review and, Nuclear Review Board review and audit requirements from TS that are and/or will be contained in the UFSAR. The changes will not alter the plant configuration (no new or different type of equipment will be installed) or create changes in methods governing normal plant operation that will introduce new failure modes. These changes will not impose different requirements and proper control of information will be maintained. These changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, these changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed TS changes do not involve a significant reduction in a margin of safety.

The proposed TS changes to revise the organization position titles, PORC composition description, and eliminate the Assistant Superintendent - Operations position, do not reduce the margin of safety because positions with equivalent authority and responsibility are established and the new positions have equivalent requirements for education, experience and training. Allowing the Plant Manager to designate appropriately qualified, trained and experienced members of the LGS staff as members of the PORC will not degrade the effectiveness of the PORC because the qualifications, training and experience level of the PORC will meet the requirements listed in ANSI/ANS 3.1-1978 and the required PORC quorum (including the use of alternates) will not be affected. Elimination of the position of Assistant Superintendent - Operations eliminates a level of supervision between the Plant Manager and the Shift Managers. If the Senior Manager - Operations does not hold an SRO license, then an Operations Manager must hold an SRO license. This individual will 1) be qualified to fill the Senior Manager - Operations position, 2) have the same management authority over the licensed operators as the Senior Manager - Operations, and 3) by being designated by Administrative procedures assures that there is always an individual holding a current SRO license in one of the Operations management positions. Other organizational changes (i.e., establishment of the positions of Manager - Operations Services and Manager - Operations Support), will ensure that the Senior Manager -Operations has sufficient time to properly supervise and monitor on-shift performance. Therefore, these changes do not involve a significant reduction in a margin of safety.

The proposed changes to the on-shift crew composition increases the number of licensed SROs per shift to be one (1) above the minimum number required by the regulations. Additionally, the title changes are consistent with the organization and reporting relationships discussed in the regulation and the LGS Updated Final Safety Analysis Report (UFSAR). The Shift Manager holds a SRO license for both units and is assigned responsibility for overall plant operation at all times when there is fuel in any unit. The other SROs on the shift report to the Shift Manager and at least one (1) of the SRO licensed individuals is in the Main Control Room when either unit is in an operating mode other than cold shutdown or refuel. The increase in the minimum on-shift crew composition and the associated changes improves the capability of the on-shift crew to safely operate the plant and SSC. Therefore, these changes do not involve a significant reduction in a margin of safety.

The proposed changes will also delete redundant and/or relocate existing independent technical review and, Nuclear Review Board review and audit requirements from TS that are and/or will be contained in the LGS UFSAR. The changes will not reduce the margin of safety since they have no impact on any safety analysis assumptions. In addition, any future changes to the UFSAR will be evaluated per the requirements of 10 CFR 50.59 or 10 CFR 50.54(a), as appropriate. Therefore, these changes will not involve a significant reduction in a margin of safety.

The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.90, to these TS details and requirements proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed changes to delete redundant and/or relocate requirements are consistent with the BWR Standard Technical Specifications (NUREG-1433) and the four criteria set forth in the NRC ''Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors,'' and since the change controls for proposed relocated details and requirements provide an equivalent level of regulatory authority, revising the TS to reflect the approved level of detail and requirements ensures no reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Attorney for licensee: J. W. Durham, Sr., Esquire, Sr. V. P. and General Counsel, Philadelphia Electric Company, 2301 Market Street, Philadelphia, Pennsylvania 19101

NRC Project Director: John F. Stolz

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: February 22, 1995

Description of amendment request: The proposed changes to the James A. Fitzpatrick Technical Specifications establish operability and surveillance requirements for the Reactor Vessel Overfill Protection Instrumentation that initiates feedwater pump turbine trips, and a main turbine trip, on high reactor vessel water level.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated because:

The proposed changes involve the addition of new operability and surveillance requirements to the Technical Specification regarding the current high reactor water level trip feature for the feedwater pump turbines and main turbine. The changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints associated with the plants instrumentation and controls. Further, the Fitzpatrick UFSAR [Updated Final Safety Analysis Report], Section 14.5.9, for the Feedwater Controller Failure operational transient does not take credit for the automatic high reactor vessel water level trip of the feedwater pump turbines. The Fitzpatrick UFSAR analysis (Section 14.5.9), for the Feedwater Controller Failure operational transient assumes an automatic high reactor vessel water level trip of the main turbine. Incorporating these requirements into the Technical Specifications provides additional assurance that a trip feature described in the UFSAR remains functional. For these reasons the changes do not increase the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from those previously evaluated because:

The proposed changes do not introduce any new accident initiators or failure mechanisms since the changes do not introduce any new modes of plant operation, make any physical changes, or alter any operational setpoints. Accordingly, the changes do not create the possibility of a new or different kind of accident from those previously evaluated.

3. Involve a significant reduction in the margin of safety because:

The proposed changes establish operability and surveillance requirements for the design feature that trips the feedwater pump turbines and main turbine on high reactor vessel water level. The requirements will assure the continued operability of a trip function that is designed to initiate protective measures in the event of excessive feedwater flow. Tripping the feedwater pump turbines and main turbine on high reactor vessel water level, precludes potential adverse safety implications associated with a reactor overfill condition. Accordingly, the proposed changes will enhance the plant safety margin.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Ledyard B. Marsh

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: March 2, 1995

Description of amendment request: The proposed changes to the James A. Fitzpatrick Technical Specifications extend the surveillance test intervals for the snubber systems to support 24 month operating cycles.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes increase the interval between snubber functional tests. These changes are consistent with the guidance provided in Generic Letter 91-04. These changes do not involve any physical changes to the plant, nor do they alter the way snubbers function. The type of testing and the actions taken if a snubber fails a functional test remain the same. The review of the snubber installation and maintenance records will continue to ensure that the snubbers service life is not exceeded prior to the next scheduled review. The proposed changes to bases 4.0 and 4.6 clarify that the snubber functional testing interval is consistent with the length of the operating cycle. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes increase the interval between snubber functional tests. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not change the ability of the snubbers to provide dynamic load support during a design basis accident. Past operating experience indicates that the snubber program at the FitzPatrick plant adequately identifies snubber failures. No changes are proposed to the type of testing performed only to the surveillance interval length. The proposed changes do not modify the design or operation of plant equipment, therefore, no new or different failure modes are introduced. The Technical Specification for snubber testing is self-corrective. If any snubber fails a functional test, Technical Specifications require additional testing of a 10% sample of that type of snubber until no more failures are found. The functional test criteria remains unchanged and ensures a 95% confidence level that at least 90% of the snubbers are operable. The proposed changes to bases 4.0 and 4.6 clarify that the snubber functional testing interval is consistent with the length of the operating cycle. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed changes increase the interval between snubber functional tests. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not alter the configuration of the snubbers nor change the manner in which the snubbers function. Operation of the facility remains unchanged by the proposed changes. An evaluation of past equipment performance indicates that snubber operability is not time dependent. The proposed changes to bases 4.0 and 4.6 clarify that the snubber functional testing interval is consistent with the length of the operating cycle. Therefore, a longer surveillance test interval will not degrade snubber performance and will not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Ledyard B. Marsh

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego County, New York

Date of amendment request: April 12, 1995

Description of amendment request: The proposed changes to the James A. FitzPatrick Technical Specifications extend the surveillance test intervals for the nuclear steam supply system to support 24 month operator cycles.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes extend the surveillance test intervals for nuclear steam supply system components. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not involve any modification to the plant, nor do they alter equipment functions. On-line testing will provide a redundant and early means of demonstrating system operability. Based on past results, SRV [safety/relief valve] mechanical performance has been good. No SRV setpoint changes are involved in this application. The proposed change to bases section 4.6 clarifies that the nuclear steam supply system surveillance testing interval is consistent with the length of the operating cycle. Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes extend the surveillance test intervals for nuclear steam supply system components. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not affect the way in which the nuclear steam supply system operates nor alter the type of surveillance testing performed. SRV drift analyses indicate that SRV drift with a 3% tolerance would be acceptable for (i.e., bounded by) a 24 to 30 month interval. Leaking or partially open SRVs are detected by the acoustic monitoring system. Since the proposed changes do not modify the design or equipment of the plant, no new failure modes are introduced. The proposed change to bases section 4.6 clarifies that the nuclear steam supply system surveillance testing interval is consistent with the length of the operating cycle. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed changes extend the surveillance test intervals for nuclear steam supply system components. These changes are consistent with the guidance provided in Generic Letter 91-04. The proposed changes do not alter the configuration of the nuclear steam supply system nor change the manner in which the system functions. Operation of the facility remains unchanged by the proposed changes. An evaluation of past equipment performance indicates that SRV mechanical performance has been good. In addition, SRV drift has been analyzed to be within the allowable tolerance for the extended surveillance interval. The proposed change to bases section 4.6 clarifies that the nuclear steam supply system surveillance testing interval is consistent with the length of the operating cycle. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mr. Charles M. Pratt, 1633 Broadway, New York, New York 10019.

NRC Project Director: Ledyard B. Marsh

Power Authority of The State of New York, Docket No. 50-286, Indian Point Nuclear Generating Unit No. 3, Westchester County, New York

Date of amendment request: March 3, 1995, as supplemented April 12, 1995

Description of amendment request: The licensee commenced operating on a 24-month fuel cycle, instead of the previous 18-month fuel cycle, with cycle 9. Fuel cycle 9 started in August 1992; however, the licensee shut down the facility in February 1993 for a performance improvement outage. Although a firm restart date has not yet been established, restart is expected in the spring of 1995. In order to accommodate operation on a 24-month cycle after the facility restarts, the licensee requested an amendment to the Technical Specifications (TSs) to incorporate the indicating instrument calibration frequency changes listed below:

(1) The licensee proposed changing the calibration frequency for the containment water level monitor instrumentation (specified in TS Table 4.1-1) to accommodate operation on a 24-month cycle.

(2) The licensee proposed changing the calibration frequency for the auxiliary feedwater (AFW) flow rate instrumentation (specified in TS Table 4.1-1) to accommodate operation on a 24-month cycle.

(3) The licensee proposed changing the calibration frequency for the containment building ambient temperature sensors (specified in TS Table 4.1-1) to accommodate operation on a 24-month cycle.

(4) The licensee proposed changing the calibration frequency for the seismic monitoring instrumentation (specified in TS Table 4.10-2) to accommodate operation on a 24-month cycle.

In addition, the licensee proposed adding a new surveillance requirement to TS Table 4.1-1 for testing the core exit thermocouples.

These proposed changes follow the guidance provided in Generic Letter 91-04, ''Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle,'' as applicable.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Consistent with the criteria of 10 CFR 50.92, the enclosed application is judged to involve no significant hazards based on the following information:

(1) Does the proposed license amendment involve a significant increase in the probability or consequences of any accident previously evaluated?

Response:

The proposed changes do not involve a significant increase in the probability or consequences of any accident previously evaluated. The proposed changes extend the calibration frequency (to 24 months) for the:

- containment temperature channels,
- containment water level monitoring system channels,
- seismic instrumentation channels, and
- auxiliary feedwater flow rate channels.

These changes are being made to accommodate a 24 month operating cycle. The proposed changes in the calibration frequencies do not involve any plant hardware changes, nor do they change the way the systems function.

Extension of the calibration and surveillance test intervals in question were evaluated and the results documented in [New York Power Authority (NYPA) Report No. IP3-RPT-MULT-00424, ''Indicating Instruments Surveillance Test Extensions,'' May 1993]. An Instrument Drift Analysis for the indicating instruments [NYPA Report No. IP3-RPT-MULT-00407, ''Instrument Drift Analysis for Indicating Loops,'' April 1993] was performed to evaluate past and future instrument drift. The results of these evaluations and analyses indicate that the calibrations in question can safely be extended to accommodate the 24 month operating cycle.

For containment temperature, auxiliary feedwater flow and seismic instrumentation, past instrument drift has generally been within acceptable limits. Some drift exceeding the calibration tolerance did occur for the triaxial time-history accelographs, but on-line testing should ensure that instrument drift over the longer cycle does not degrade system performance. For containment water level systems (except containment building level), new electronic transmitters were recently installed. Due to the lack of data, an instrument drift analysis was not performed. However, the new containment water level transmitters improved the overall channel accuracy.

Future instrument drift was predicted and used to update existing loop accuracy calculations, with the following results. (1) For the containment temperature channels, the loop accuracy calculations were revised to incorporate the larger channel uncertainties. Postulated drift over 30 months should have a negligible effect on the EOPs [Emergency Operating Procedures] and plant shutdown. (2) For the containment system sump water levels, future drift is not a concern because the containment building water level is used post accident. The larger uncertainties can safely be accommodated by changing the EOP setpoint for transfer to cold leg recirculation. (3) For the seismic instrumentation, past drift was negligible, and future drift is not expected to be cycle length dependent. (4) For the auxiliary feedwater flow rate channels, the larger uncertainties can be safely accommodated by changing the EOP setting for the minimum AFW flow required for heat removal.

For the containment temperature and seismic instrumentation, on-line testing provides added assurance that the instrumentation is functioning as required.

[For the core exit thermocouples, adding a requirement to conduct testing every 18 months will serve to ensure system operability. This new testing requirement does not change the way the plant operates or involve hardware modifications.]

(2) Does the proposed license amendment create the possibility of a new or different kind of accident from any previously evaluated?

Response:

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated. The proposed changes extend the calibration frequency (to 24 months) for the:

• containment temperature channels,
• containment water level monitoring system channels,
• seismic instrumentation channels, and
• auxiliary feedwater flow rate channels.

These changes are being made to accommodate a 24 month operating cycle. The proposed changes in the calibration frequencies do not involve any plant hardware changes, nor do they change the way the systems function.

Extension of the calibration and surveillance test intervals in question were evaluated and the results documented in [same as Question (1)]. An Instrument Drift Analysis for the indicating instruments [same as Question (1)] was performed to evaluate past and future instrument drift. The results of these evaluations and analyses indicate that the calibrations in question can safely be extended to accommodate the 24 month operating cycle. For the containment temperature and seismic instrumentation, on-line testing provides added assurance that the instrumentation is functioning as required.

[For the core exit thermocouples, adding a requirement to conduct testing every 18 months will serve to ensure system operability. This new testing requirement does not change the way the plant operates or involve hardware modifications.]

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response:

The proposed changes do not involve a significant reduction in a margin of safety. The proposed changes extend the calibration frequency (to 24 months) for the:

• containment temperature channels,
• containment water level monitoring system channels,
• seismic instrumentation channels, and
• auxiliary feedwater flow rate channels.

These changes are being made to accommodate a 24 month operating cycle. The proposed changes in the calibration frequencies do not involve any plant hardware changes, nor do they change the way the systems function.

For containment temperature, auxiliary feedwater flow and seismic instrumentation, past instrument drift has generally been within acceptable limits. Some drift exceeding the calibration tolerance did occur for the triaxial time-history accelographs, but on-line testing should ensure that instrument drift over the longer cycle does not degrade system performance. For containment water level systems (except containment building level), new electronic transmitters were recently installed. Due to the lack of data, an instrument drift analysis was not performed. However, the new containment water level transmitters improved the overall channel accuracy.

[For the core exit thermocouples, adding a requirement to conduct testing every 18 months will serve to ensure system operability. This new testing requirement does not change the way the plant operates or involve hardware modifications.]

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10601.

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: Ledyard B. Marsh

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of amendment request: March 30, 1995

Description of amendment request: The proposed change to the Technical Specifications eliminates the defined term CONTROLLED LEAKAGE, removes Controlled Leakage flow from the Reactor Coolant System Operational Leakage Limiting Condition for Operation (LCO), and establishes a new Seal Injection Flow LCO.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Do not involve a significant increase in the probability or consequence of an accident previously evaluated.

Changing the Technical Specification to limit seal injection flow instead of seal leakoff flow does not affect the probability of any accident previously evaluated. Maintaining adequate Emergency Core Cooling System (ECCS) flow during Loss of Coolant Accident (LOCA) ensures that the consequences of these accidents are unaffected. The existing Technical Specification allows seal injection throttle valve positioning that could result in seal injection flow path resistance values below those used in the Salem ECCS hydraulic flow analyses. Reduced line resistances could result in inadequate ECCS flow to the reactor core. Revising the Technical Specification to limit RCP seal injection flow ensures that the accident analysis assumptions are maintained, and the previously evaluated accident consequences remain unchanged.

Therefore, it may be concluded that the proposed changes do not increase the probability or consequences of an accident previously evaluated.

2. Do not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve any hardware modifications or result in any functional changes to system operation. RCP seal injection flow is used as a limiting parameter in-place of RCP seal leakoff flow.

Since design requirements continue to be met and the RCS pressure boundary is not challenged, no new failure mode is created. Thus, an accident different from any already evaluated is not created by this change.

Therefore, it may be concluded that the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do not involve a significant reduction in a margin of safety.

The proposed changes do not alter the manner in which Safety Limits or Limiting Safety System Setpoints are determined. Controlled Leakage (RCP seal leakoff)is removed from the Reactor Coolant System Leakage Limiting Condition for Operation (LCO), and a new seal injection LCO is established. The new LCO continues to limit seal injection flow during accident conditions. The limiting parameter is changed from RCP seal leakoff flow to RCP seal injection flow. These changes ensure that the accident analysis assumptions and existing margins of safety are maintained. The seal injection flow specification limit is not applicable in Mode 4 and lower, because high seal injection flow is less critical due to lower Reactor Coolant System (RCS) pressure and decay heat removal requirements in these modes. Reactor coolant pump seal injection flow must be limited in Modes 1, 2, and 3 to ensure adequate Emergency Core Cooling System Flow.

Therefore, it may be concluded that the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Salem Free Public library, 112 West Broadway, Salem, New Jersey 08079

Attorney for licensee: Mark J. Wetterhahn, Esquire, Winston and

Strawn, 1400 L Street, NW, Washington, DC 20005-3502

NRC Project Director: John F. Stolz

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: April 6, 1995 (TS 95-05)

Description of amendment request: The proposed change would (1) replace the reference to Table 3.6-2 from Definition 1.7.a.2 for Containment Integrity with a phrase that will allow the valves to be opened under administrative control; (2) replace the reference to Table 3.6-2 from Surveillance Requirement 4.6.1.1 with a phrase that will allow the valves to be opened under administrative control; (3) delete the reference to Table 3.6-1 from Technical Specification 3.6.1.2; (4) delete Table 3.6-1, ''Bypass Leakage Paths to the Auxiliary Building -- Secondary Containment Bypass Leakage Paths;'' (5) revise Specification 3.6.3 to delete the reference to Table 3.6-2, add a footnote that discusses the opening of penetrations intermittently, add the phrase to take exception to the containment vacuum isolation valves, and add an action statement to indicate that Specification 3.0.4 does not apply to the specification; (6) delete Surveillance Requirement 4.6.3.1; (7) delete references to Table 3.6-2 in Specifications 4.6.3.2 and 4.6.3.3 and additional wording added to indicate that the specifications apply to automatic containment isolation valves; (8) delete Table 3.6-2, ''Containment Isolation Valves'' and add a note to the page indicated that the information has been intentionally deleted; (9) revise Specification 3.8.3.1 to specify that the Limiting Condition for Operation applies to primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration shall be operable, add a phrase to indicate that the scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating, and delete the phrase that references appropriate plant instructions in the action statement; (10) delete the phrase that references appropriate plant procedures from Specification 4.8.3.1; (11) delete the phrase from SR 4.8.3.1.a.3 that indicates that a complete listing of all fuses to be verified in accordance with the requirement will be maintained in appropriate plant instructions; (12) replace the phrase ''appropriate plant instructions based on'' with ''procedures prepared in conjunction with'' in SR 4.8.3.1.b; (13) replace the reference to Table 3.8-2 in Specification 3.8.3.2 with a phrase that indicates that the Requirement is applicable to valves used in safety systems; (14) delete Table 3.8-2, ''Motor Operated Valves Thermal Overload Protection,'' and replace it with a note that indicates that the pages are intentionally blank; and (15) incorporate appropriate changes to the Bases to reflect these changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

TVA has evaluated the proposed technical specification (TS) change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of Sequoyah Nuclear Plant (SQN) in accordance with the proposed amendment will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The removal of the component listings from the SQN TSs will not create an increase in the probability or consequences of any accident previously evaluated. Although no longer in the TSs, the components listed in Tables 3.6-1, 3.6-2, and 3.8-2 will be contained in administratively controlled documents. This equipment must be tested at the required intervals and each unit's action statements must still be adhered to. These procedures are revised and approved in accordance with requirements of TS Section 6.5.1A. This review process also requires an evaluation based on 10 CFR 50.59 requirements. As indicated in GL 91-08, this is adequate control for changes to these components lists.

2. Create the possibility of a new or different kind of accident from any previously analyzed.

The removal of the component lists from the TSs does not modify safety-related equipment or systems, nor does it change any safety-related setpoints used to prevent or mitigate previously analyzed accidents. The component lists are presently located in separate documents that are subject to the requirements of 10 CFR 50.59. Also, the limiting condition of operation requirements remain in effect and appropriate actions will be taken if any limits are exceeded. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The margin of safety is not affected by the removal of the previously discussed component lists from the TS. Appropriate measures presently exist to control the setpoint of the components listed. Any changes to these setpoints are controlled by the SQN design change process that is subject to the requirements of 10 CFR 50.59 in which the reduction of the present margin of safety is addressed. The proposed amendment continues to require operation within the set values for these components, and appropriate actions to be taken when or if the limits are exceeded. Based on these controls, this amendment will not involve a reduction in a margin of safety.

The NRC has reviewed the licensee's analysis and, based on thisreview, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Chattanooga-Hamilton County Library, 1101 Broad Street, Chattanooga, Tennessee 37402

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11H, Knoxville, Tennessee 37902

NRC Project Director: Frederick J. Hebdon

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: March 24, 1995

Description of amendment request: The licensee has requested a one-time extension of the performance intervals for certain Technical Specification Surveillance Requirements (SR). Affected SRs include penetration leak rate testing, valve operability testing, instrument calibration, response time testing, and logic system functional tests. The proposed changes are requested to support refueling outage 5 scheduled to begin no later than February 15, 1996.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed TS change requests a one-time extension of the surveillance intervals related to: a) RPS Instrumentation calibration, LSFTs, and response time testing; b) Isolation Actuation System Instrumentation calibration, LSFTs, and response time testing; c) ECCS Actuation Instrumentation calibration, LSFTs, and response time testing; d) Control Rod Block Instrumentation calibration and LSFTs; e) Remote Shutdown Instrumentation and Controls calibration and operability testing; f)

Accident Monitoring Instrumentation calibration; g) Plant Systems Instrumentation calibration and LSFTs; h) Primary Containment automatic valve actuation; i) Reactor Coolant System Pressure Isolation Valve (PIV) testing; j) system automatic initiation testing; and, k) Emergency Diesel Generator inspection and testing.

Also proposed is the re-establishment of the baseline for the ''N times 18 months'' cumulative surveillance interval for response time testing.

The discussion in the License Amendment Request demonstrates the following:

i) Rosemount transmitter calibration period extension is acceptable based on Rosemount D8900126, Revision A which supported extension of the calibration interval from 18 months to 30 months based on the reduction in the drift allowance;

ii) Extrapolation of plant specific calibration data is acceptable in supporting the extension of other calibration surveillance intervals to RFO-5;

iii) LSFT interval extension is acceptable based on the NRC Safety Evaluation Report (Peach Bottom Atomic Power Plant, Units 2 and 3, dated August 2, 1993) which supported extension of the interval for LSFT from 18 to 24 months. This was based on the small probability of relay or contact failure relative to mechanical component failure probability and, therefore, the increase in LSFT interval represented no significant change in the overall safety system unavailability;

iv) Response time testing interval extension for Isolation Actuation and ECCS Actuation instrumentation channels is acceptable based on the BWR Owners Group (BWROG) Licensing Topical Report NEDO-32291 (January 1994) which provided the necessary justification for elimination of response time testing and, therefore, provides a suitable argument for extending the interval for a short period of time. The NRC approved the use of NEDO-32291 as a basis for License Amendment Requests, with additional conditions specified, in a letter to the BWROG in December 1994.

v) Response time testing interval extension for RPS Instrumentation channels is acceptable because: i) there are redundant sensors that can initiate the scram function; ii) one-out-of-two redundancy exists in every individual instrument channel within each trip function; iii) several redundant and diverse instrument channels are provided which can detect and generate a scram signal; iv) the failure probability is a small fraction of the total control rod insertion (scram) failure probability; v) failure of instrumentation in the sluggish mode is a small fraction of its overall failure modes; and iv) NRC Safety Evalution Report dated August 2, 1993 (Peach Bottom Atomic Power Station, Units 2 and 3 docket) has previously provided approval for extension of the RPS response time testing surveillance interval from 18 to 24 months.

vi) Response time testing interval extension for the Main Steam Line isolation is acceptable because i) redundancy and diversity exist in individual instrument channels within a trip function; ii) instrumentation response time is a small fraction of the overall response time of the actuating device; iii) instrumentation failure probability is a very small portion of the total MSIV failure probability; and, iv) failure of instrumentation in the sluggish responding mode is a small fraction of its overall failure modes.

vii) Containment Isolation Valve leakage determination and actuation interval extension is acceptable based on: i) redundancy provided in the design of the penetrations; ii) the periodic testing of the valves during power operation; and, iii) the short period of time the interval is being extended.

viii) Reactor Coolant System PIVs have exhibited low as-found leak rates as measured during the last refueling outage; there is substantial margin available for the PIVs from the as-left leakage to the allowed TS leakage; the requested extension of the surveillance interval is small; and the conclusion of NUREG-1463, ''Regulatory Analysis for the Resolution of Generic Safety Issue 105: Interfacing System Loss-of-Coolant Accident in Light Water Reactors'' (July 1993), and the confirmation of the PNPP Individual Plant Examination that the ISLOCA (for which PIVs are provided to prevent) is not a risk concern to BWRs or PNPP.

ix) System initiation and actuation testing interval is acceptable based on the periodic testing of components during power operation and the short period of time the interval is being extended.

x) Emergency Diesel Generator testing interval extension is acceptable based on: i) the past testing results which support extension for the short period of time; ii) the testing that is done during power operation; and, iii) the short period of time the interval is being extended.

xi) The re-establishment of the baseline for the ''N times 18 months'' cumulative surveillance interval for response time testing is acceptable in that the extension of the cumulative interval would not be for more than the individual extension requested and justified herein.

Therefore, from the above it is shown that the proposed change will not significantly increase the probability of an accident previously evaluated.

2. The proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed TS change requests a one-time extension of the surveillance intervals for instrument calibration, instrument channel LSFT and response time testing, containment isolation valve leakage determination and actuation, PIV leak rate determination, system actuation testing, and diesel generator inspection and testing. The proposed changes do not necessitate a physical alteration to the plant (no new or different type of equipment will be installed). The requested extension durations are small as compared to the overall interval allowed by TS; drift data supports extension of the calibration intervals; NRC and industry evaluations support extension of LSFT; industry evaluations and redundancy in system design support extension of response time testing; past testing and periodic testing provides confidence of no effect on equipment availability by extending the confidence of no effect on equipment availability by extending the surveillance interval. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

In addition, the requested re-establishment of the baseline at RFO-5 for the ''N time 18 months'' cumulative surveillance interval for response time testing is acceptable in that the cumulative surveillance interval will not be extended by more than that which is proposed for individual response time tests during RFO-5. The individual response time test surveillance interval extensions have been justified herein. The justification for individual response time test surveillance interval extensions applies to the cumulative surveillance interval extension which is requested and will be granted by allowing the re-establishment of the baseline of the ''N times 18 months'' surveillance interval to the response time testing dates for those response time tests to be performed during RFO-5. The proposed changes do not necessitate a physical alteration to the plant (no new or different type of equipment will be installed). Therefore, the change does not create the possibility of a new or different kind of accident.

3. The proposed change will not involve a significant reduction in the margin of safety.

The proposed TS change requests a one-time extension of the surveillance intervals for instrument calibration, instrument channel LSFT, and response time testing, containment isolation valve leakage determination and actuation, PIV leak rate determination, system actuation testing, and diesel generator inspection and testing. The proposed changes do not necessitate a physical alteration to the plant (no new or different type of equipment will be installed). In that the requested extension durations are small as compared to the overall interval allowed by TS, drift data supports extension of the calibration intervals, NRC and industry evaluations support extension of LSFT, industry evaluations and redundancy in system design support extension of response time testing, past testing and periodic testing provides confidence of no effect on equipment availability by extending the surveillance interval, the change does not involve a significant reduction in the margin of safety.

In addition, the requested re-establishment of the baseline at RFO-5 for the ''N times 18 months'' cumulative surveillance interval for response time testing is acceptable in that the cumulative surveillance interval will not be extended by more than that which is proposed for individual response time tests during RFO-5. The individual response time test surveillance interval extensions have been justified herein. The justification for individual response time test surveillance interval extensions applies to the cumulative surveillance interval extension which is requested and will be granted by allowing the re-establishment of the baseline of the ''N times 18 months'' surveillance interval to the response time testing dates for those response time tests to be performed during RFO-5. The proposed changes do not necessitate a physical alteration to the plant (no new or different type of equipment will be installed). Therefore, the change does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Gail H. Marcus

The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, Toledo Edison Company, Docket No. 50-440, Perry Nuclear Power Plant, Unit No. 1, Lake County, Ohio

Date of amendment request: April 3, 1995

Description of amendment request: The proposed amendment would add new programmatic requirements governing radiological effluent into the Administrative Controls section of the Technical Specifications in accordance with Generic Letter 89-01, ''Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program.''

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes are administrative in nature and alter only the format and location of programmatic controls and procedural details relative to radioactive effluent, radiological environmental monitoring, solid radioactive wastes, and associated reporting requirements. Compliance with applicable regulatory requirements will continue to be maintained. In addition, the proposed changes do not alter the conditions or assumptions in any of the Updated Safety Analysis Report (USAR) accident analyses. Since the USAR accident analyses remain bounding, the radiological consequences previously evaluated are not adversely affected by the proposed changes. Therefore, it can be concluded that the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

The proposed changes do not involve any changes to the configuration or method of operation of any plant equipment. Accordingly, no new failure modes have been defined for any plant system or component important to safety nor has any new limiting single failure been identified as a result of the proposed changes. Also, there will be no change in types or increase in the amounts of any radioactive effluent released offsite. Therefore, it can be concluded that the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety.

The proposed changes do not involve any actual change in the methodology used in the control of radioactive effluents, solid radioactive wastes, or radiological environmental monitoring. These changes are considered administrative in nature, provide for the relocation of procedural details outside the Technical Specifications, and add appropriate administrative controls in the Technical Specifications to provide continued assurance of compliance with applicable regulatory requirements. These proposed changes also comply with the guidance contained in Generic Letter 89-01. Therefore, it can be concluded that the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Perry Public Library, 3753 Main Street, Perry, Ohio 44081

Attorney for licensee: Jay Silberg, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, NW., Washington, DC 20037

NRC Project Director: Gail H. Marcus

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: February 24, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Surveillance Requirement 4.6.1.7.4 and its associated Bases to delete the quarterly verification of the measured leakage rate for containment mini-purge supply and exhaust isolation valves.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed revision does not involve a significant hazards consideration because operation of Callaway Plant with this change would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed revision to the T/S will not adversely impact plant safety since the requirement to perform the quarterly surveillance will still be implemented to verify valve leakage and seal degradation. The mini-purge valves will still perform their intended safety function to close within 5 seconds after receipt of an isolation signal.

2) Create the possibility of a new or different kind of accident from any previously evaluated.

There are no design changes being made that would create a new type of accident or malfunction and the method and manner of plant operation remain unchanged. Deletion of the individual leakage rate for these valves does not affect the severity of any accident previously evaluated. The consequences of a valve failure or malfunction are not increased by the removal of the acceptance criteria, leakage rate will still be measured on a quarterly basis as is currently done to determine if the seals are degrading.

3) Involve a significant reduction in a margin of safety.

There are no changes being made to the safety limits or safety system settings that would adversely impact plant safety. The valves will still be surveilled on a quarterly basis to verify leakage and seal degradation to assure gross failure will not occur and that containment integrity is maintained.

Based on the above discussions, it has been determined that the requested Technical Specification change does not involve a significant increase in the probability or consequences of an accident or create the possibility of a new or different kind of accident or condition over previous evaluations; or involve a significant reduction in a margin of safety. Therefore, the requested license amendment does not involve a significant hazards consideration.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, DC 20037

NRC Project Director: Gail H. Marcus

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of amendment request: April 17, 1995

Description of amendment request: The proposed amendment would revise Technical Specification (TS) Table 2.2-1 and associated Bases to reduce repeated alarms and partial reactor trips related to the C-4 control system interlock and the Overpower Delta-T (OP[delta]T) reactor trip setpoint.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

The proposed revision does not involve a significant hazards consideration because operation of Callaway Plant with this change would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

Overall protection system performance will remain within the bounds of the accident analyses documented in Final Safety Analyses Report (FSAR) Chapter 15, WCAP-10961-P for Category 1 plants such as Callaway, and WCAP-11883 since no hardware changes are proposed.

The OP[delta]T reactor trip function provides protection against excessive power (fuel rod integrity protection within the fuel temperature design basis). No credit is taken for the OP[delta]T trip in the Chapter 15 licensing basis accident analyses. The [delta]T trip function is credited in non-licensing basis analyses of various steamline breaks.

The OP[delta]T trip will continue to function in a manner consistent with the plant design basis. There will be no change to the OP[delta]T safety analysis limit listed in FSAR Table 15.0-4. Therefore, there will be no degradation in the performance of or an increase in the number of challenges to equipment assumed to function during an accident situation.

The reactor trip system response time, as defined in the Technical Specifications, will be unaffected.

These Technical Specification revisions do not involve any hardware changes nor do they affect the probability of any event initiators. There will be no change to normal plant operating parameters or accident mitigation capabilities. Therefore, these changes will not increase the probability or consequences of an accident or malfunction.

2) Create the possibility of a new or different kind of accident from any previously evaluated.

As discussed above, there are no hardware changes associated with these Technical Specification revisions nor are there any changes in the method by which any safety-related plant system performs its safety function. Revisions to the OP[delta]T values for K_4 and K_6 will require scaling changes for summing amplifier cards (NSA cards) in the 7300 Process Protection System. These scaling changes are straightforward and similar in nature to those performed to implement OL Amendments 72 and 84 associated with the implementation of relaxed axial offset control (RAOC) and a revised OT[delta]T f_1([delta]I) penalty function. These scaling changes will not affect the normal manner of plant operation. There will be a reduction in the incidence of C-4 alarms and partial reactor trips. There will be less of a need to reduce power during on-line surveillance testing.

No new accident scenarios, transient precursors, failure mechanisms, or limiting single failures are introduced as a result of these changes. There will be no adverse effect or challenges imposed on any safety-related system as a result of these changes. Therefore, the possibility of a new or different kind of accident is not created.

3) Involve a significant reduction in a margin of safety.

There will be no change to the Overpower [delta]T safety analysis limit listed in FSAR Table 15.0-4. Available setpoint calculation margin will be used to increase the K_4 value, reflected as a new bias on a summing amplifier card in each of the four protection loops. This will also require corresponding decreases in the OP[delta]T Total Allowance and Allowable Value in Technical Specification Table 2.2-1. Available margin in the OP[delta]T trip protection function will be used to decrease the K_6 value, reflected as a new gain on a summing amplifier card in each of the four protection loops.

As discussed above, the response time of the OP[delta]T reactor trip function will remain unchanged.

It has been confirmed that the Z and S terms currently listed in Table 2.2-1 for the OP[delta]T trip function will remain conservative. The change in K_4 will result in a decrease in the Total Allowance and Allowable Value for OP[delta]T; however, this does not affect any margin of safety since the safety analysis limit, which preserves the overpower safety margin, is unchanged.

There will be no effect on the manner in which safety limits or limiting safety system settings are determined nor will there be any effect on those plant systems necessary to assure the accomplishment of protection functions. There will be no impact on the overpower limit, DNBR limits, F_Q, F[delta]H, LOCA PCT, peak local power density, or any other margin of safety.

Based upon the preceding information, it has been determined that the proposed changes to the Technical Specifications do not involve a significant increase in the probability or consequences of an accident previously evaluated, create the possibility of a new or different kind of accident from any accident previously evaluated, or involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts & Trowbridge, 2300 N Street, N.W., Washington, DC 20037

NRC Project Director: Gail H. Marcus

Vermont Yankee Nuclear Power Corporation, Docket No. 50-271, Vermont Yankee Nuclear Power Station, Vernon, Vermont

Date of amendment request: October 28, 1994

Description of amendment request: The proposed amendment would remove the Neutron Monitoring System (NMS) and Control Rod Position instrumentation from the Vermont Yankee Technical Specifications for post-accident monitoring. Administrative changes are also proposed.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change to remove the NMS and Control Rod Position instrumentation from the Technical Specifications for post-accident monitoring is consistent with NRC requirements concerning this instrumentation.

Wide Range Neutron Flux (NMS instrumentation) is presently included in the [boiling water reactor] BWR Standard Technical Specifications, but the NRC has recently determined [letter, USNRC to VYNPC, dated April 29, 1993] that this instrumentation need not meet R.G. 1.97 Category 1 criteria and that licensees may request the removal of this instrumentation from their post-accident monitoring Technical Specifications. Control Rod Position instrumentation is considered R.G. 1.97 Category 3 which is required to meet the least stringent design and qualification criteria as specified in this regulatory guide.

Testing, calibration and maintenance of this instrumentation will continue to assure operability of instrumentation. The portions of the NMS and the Control Rod Position instrumentation systems to be removed from the post-accident monitoring Technical Specifications do not perform any automatic control or trip function. In addition, this instrumentation does not provide information that is required to permit the control room operator to take manual actions that are required for safety systems to accomplish their safety functions for design basis accident events.

At a BWR, when all control rods are inserted, these control rods cannot be withdrawn without deliberate operator action. The proposed change does not result in any system hardware modification or new plant configuration. The requested change to post-accident monitoring instrumentation does not impact any [Final Safety Analysis Report] FSAR safety analysis involving the NMS or Control Rod Position System. These monitoring functions are not contributors to the initiation of accidents.

The administrative changes to correct a typographical error and instrument ranges will have no effect on plant hardware, plant design, safety limit setting or plant system operation and therefore, do not modify or add any initiating parameters that would significantly increase the probability or consequences of any previously analyzed accident.

Therefore, it is concluded that there is not a significant increase in the probability or consequences of an accident previously evaluated.

2. The function of the instrumentation to be removed from the Technical Specifications is for monitoring only. These indications are not necessary for operators to accomplish any safety functions.

The proposed change does not involve any change in hardware, Technical Specification setpoints, plant operation, redundancy, protective function or design basis of the plant. There is no impact on any existing safety analysis or safety design limits. NMS and Control Rod Position monitoring functions do not initiate nuclear system parameter variations which are considered potential initiating causes of threats to the fuel and the nuclear system process barrier.

As discussed above, the proposed administrative change only corrects a typographical error concerning equipment identification numbers and listed instrument ranges. This change does not affect any equipment and they do not involve any potential initiating events that would create any new or different kind of accident.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change to remove the NMS and Control Rod Position instrumentation from the Technical Specifications for post-accident monitoring does not affect any existing safety margins. The original NMS design basis for BWRs never required a post-accident neutron monitoring function since there are no design basis accidents that rely on operator action to control reactor power. This is also true for Control Rod Position monitoring.

Existing Technical Specifications requirements for automatic trip functions are unaffected. Failure of the indication of reactor power from the NMS or the Control Rod Position System does not preclude the ability of the reactor operator to determine reactor power levels. Alternate indications are available to ascertain reactor power. These include reactor coolant boron concentrations, flux levels from the Traversing Incore Probe (TIP) System and the status of plant parameters which are linked to reactor power. In addition, alternate means of determining reactor power have been incorporated into the Emergency Operating Procedures (EOPs).

Operation, testing and maintenance of this instrumentation will remain the same. System functions are the same. Post-accident functional design criteria as described in [BWR Owners Group Topical Report NEDO-31558-A, dated March 29, 1993], and approved by the NRC are satisfied by present equipment installed at VY. NMS instrumentation is still included in the Technical Specifications for the [Reactor Protection System] RPS. Control Rod Position instrumentation does not perform any safety function.

As discussed above, the proposed administrative changes do not affect any equipment involved in potential initiating events or safety limits.

Based upon the above, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above, we conclude that the proposed change does not constitute a significant hazards consideration as defined in 10CFR50.92(c).

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Brooks Memorial Library, 224 Main Street, Brattleboro, VT 05301

Attorney for licensee: John A. Ritsher, Esquire, Ropes and Gray, One International Place, Boston, MA 02110-2624

NRC Project Director: Phillip F. McKee

Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: March 30, 1995

Description of amendment request: The licensee is requesting temporary changes to Technical Specifications (TS) 3.7.3.1, ''Component Cooling Water Subsystem - Operating,'' and 3.7.4.1, ''Service Water System - Operating,'' for NA-1&2. The proposed TS changes will allow one of the two service water loops to be isolated from the component cooling water heat exchangers during power operation in order to refurbish the isolated service water headers.

NA-1&2 is currently pursuing refurbishment of the 18-inch, 20-inch and 24-inch diameter service water supply and return lines to/from the NA-1 and NA-2 component cooling heat exchangers (CCHXs). Refurbishment of this piping presents a challenge in that it is not possible to isolate and plug or blank the section to be worked in a 7-day time period. The purpose of the proposed change is to request temporary changes to the existing servicewater (SW) and component cooling water (CC) TS to permit orderly and efficient conduct of the pipe refurbishment project during two-unit power operation. Specifically, the licensee is proposing to temporarily change TS 3.7.4.1 ''Service Water System - Operating'' to allow operation of the SW system with one independent source of SW to/from the NA-1 and NA-2 CCHXs for two periods of up to 49 days each. This proposed change also allows the automatic closure feature of the SW valves to/from the CCHXs to be defeated during the 49-day periods. In addition, the licensee proposes to temporarily change TS 3.7.3.1 ''Component Cooling Water Subsystem - Operating'' with a footnote which considers the CC subsystems OPERABLE with only one independent source of SW provided to/from the CCHXs during these 49-day periods. Further, the proposed change would allow that during operation with only one SW header available to/from the CCHXs, the provisions of Specification 3.0.4 would not be applicable provided two SW loops are capable of providing cooling for the other operable plant components.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Specifically, operation of North Anna Power Station in accordance with the proposed Technical Specifications changes will not:

Involve a significant increase in the probability or consequences of an accident previously evaluated.

The piping refurbishment project and the proposed temporary changes to the SW and CC Technical Specifications have been evaluated to assess their impact on the normal operation of the SW and CC systems and to ensure that the design basis safety functions of each system are preserved. The SW system is required to function during all normal and emergency operating conditions. During normal plant operation, the SW system provides cooling water to the CCHXs, charging pump coolers, instrument air compressor coolers, and control room chiller condensors of both units. During the two 49-day periods, one header will [operate] with its 24-inch piping to/from the CCHXs temporarily blanked. To avoid operation of the SW pump at abnormal conditions (low flow) on this ''partially deadlocked'' header, a temporary cross-connect will be installed to by-pass the CCHXs.

SW system operation with the cross-connect installed was evaluated for design basis accident (DBA) conditions. The DBA condition for the SW system is a loss-of-coolant accident on one unit with simultaneous loss-of-offsite-power to both units. A SW system hydraulic analysis has been performed to verify that adequate flow is provided to the containment recirculation spray heat exchangers (RSHXs) with the temporary cross-connect installed and throttled open assuming the occurrence of the most limiting single failure. Therefore, there is no increase in probability or consequences of the DBA condition.

Utilizing only one SW header to supply flow to the CCHXs has the potential to affect the reliability of the CC system and all of the equipment cooled by CC. The activities to be performed during the refurbishment project and the various system alignments required have been evaluated using the Individual Plant Examination (IPE) Probabilistic Safety Assessment (PSA) model for North Anna Power Station. This model is used in a manner that is generally consistent with the Nuclear Energy Institute (NEI)/Electric Power Research Institute (EPRI) draft PSA Applications Guide (Revision H). The effect on the PSA model is a slight increase in the frequency of reactor trips and an increase in the probability of RHR failure.

The increased frequency of reactor trips is due to the decreased reliability of the CC system to supply cooling to the reactor coolant pump (RCP) motors. When only one SW header is available to the CCHXs, the increased frequency of losing this single header can be conservatively estimated by combining the failure probability of both SW pumps (approximately 1.5E-4 based on IPE PSA data). Also considered was the frequency of pipe rupture anywhere in the single available header. When the single SW header fails to supply cooling to the CCHXs, the CC system will heatup causing inadequate cooling for sustained operation of the RCPs. Tripping these pumps results in a reactor trip. The second SW header can be expected to supply other equipment with cooling. A sensitivity analysis shows the increase in CDF as a result of the increased reactor trip frequency to be less than 1E-8 per year.

The CC system is also included in the PSA model as a support system for RHR cooling. The RHR system is used to reduce reactor coolant system temperatures from 350°F (hot shutdown) to 140°F (cold shutdown). The only accident initiator that requires the unit to be cooled down and placed on RHR cooling are sequences which are initiated with a steam generator tube rupture. (Note that, for the North Anna plant design, RHR is separate from the safety injection system and the low head safety injection pumps.) The increased probability for the loss of RHR when only one SW header is available to the CCHXs is estimated using fault tree analysis and is dominated by the failure of both SW pumps. The probability for the loss of both SW pumps aligned to the CCHXs is estimated to be 1.5E-4. The effect of this increase in RHR failure probability was determined by adding this probability to the top single event in the RHR function and recalculating the new CDF. The resulting increase in CDF as a result of RHR system failure following a steam generator tube rupture is less than 1E-8 per year.

The CC system is further included in the PSA model as part of the loss of RCP seal cooling as an initiating event and as a loss of function during other initiating event scenarios. The effect on the probability for a loss of RCP seal cooling due to losing CC cooling to the RCP thermal barriers is negligible due to the high reliability of the charging system to provide seal injection.

The total effect of this pipe refurbishment project was estimated by a sensitivity analysis combining both the change in the reactor trip initiating event frequency and the increased failure probability of RHR resulting in less than a 1E-6 per year increase in CDF. Since this project will not affect the containment systems, there would not be any significant change in off-site dose, except that resulting directly from the increase in CDF. These minor increases in CDF and off-site dose are less than what is defined as risk significant in the NEI/EPRI draft PSA Applications Guide.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed temporary Technical Specifications changes do not affect the basic method of operation of the SW or CC systems. The purpose of the proposed changes is to permit extended operation of the CC system with one independent source of SW cooling. During the project, there will be a significant time period when all the CCHXs are aligned to one SW loop, the possibility of an interruption of SW supply to the heat exchangers during a DBA is eliminated by defeating the closure of the 24-inch SW isolation MOVs to the CCHXs on a SI/CDA signal. Both SW headers will be available for equipment required for safe shutdown of the units (i.e., RSHXs, charging pumps, and CR/ESGR chillers). The SW pipe repair activities and the installation/removal of the SW cross-connect piping do not create the possibility for a malfunction of equipment different than previously evaluated. Therefore, implementation of the restoration project and approval of the proposed Technical Specifications changes will not introduce any new accident initiators nor affect the performance of accident mitigation systems.

3. Involve a significant reduction in a margin of safety.

The proposed changes to the schedule only provide operational flexibility to perform the required SW pipe refurbishment. The Technical Specifications continue to require the SW and CC systems to remain functional during the period with a single SW supply to the CCHXs. As stated in item (1) above, the SW system is fully capable of performing its DBA function during the course of the pipe refurbishment project with the proposed Technical Specification changes in place. The effect of this pipe refurbishment project on CC system reliability was estimated by a sensitivity analysis combining both the change in the reactor trip initiating event frequency and the increased failure probability of RHR resulting in less than a 1E-6 per year increase in CDF. Since this project will not affect the containment systems, there would not be any significant change in off-site dose, except that resulting directly from the increase in CDF. These minor increases in CDF and off-site dose are less than what is defined as risk significant in the NEI/EPRI draft PSA Applications Guide. Therefore, there is not a significant reduction in margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: The Alderman Library, Special Collections Department, University of Virginia, Charlottesville, Virginia 22903-2498.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Riverfront Plaza, East Tower, 951 E. Byrd Street, Richmond, Virginia 23219.

NRC Project Director: David B. Matthews

Previously Published Notices Of Consideration Of Issuance Of Amendments To Facility Operating Licenses, Proposed No Significant Hazards Consideration Determination, And Opportunity For A Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the **Federal Register** on the day and page cited. This notice does not extend the notice period of the original notice.

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: March 31, 1995

Brief description of amendments: The proposed amendments would provide an exception to Technical Specification (TS) 3.0.4. TS 3.0.4 allows entry of a unit into another operational condition only if the conditions of the Limiting Conditions for Operation (LCOs) are met without reliance on TS action statements. The exception requested by the licensee would allow a change in a unit's operational condition in a specific situation in which the unit's LCO concerning the minimum number of operable offsite power circuits is not fully satisfied. Specifically, the exception would allow an operational mode change of a unit if the second unit is in Operational Condition 4 or 5 (i.e., cold shutdown or refueling) and one of the second unit's offsite power circuits is inoperable.

Date of publication of individual notice in **Federal Register:** April 13, 1995 (60 FR 18860)

Expiration date of individual notice: May 15, 1995

Local Public Document Room location: The University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297

Duquesne Light Company, et al., Docket No. 50-412, Beaver Valley Power Station, Unit No. 2, Shippingport, Pennsylvania

Date of amendment request: April 10, 1995, as supplemented April 12, 1995

Brief description of amendment request: The proposed amendment would revise Technical Specification (TS) 4.6.2.2.d to delete the reference to the specific test acceptance criteria for the Containment Recirculation Spray Pumps and replace the specific test acceptance criteria with reference to the requirements of the Inservice Testing (IST) Program. In addition, the 18-month test frequency would be replaced with the test frequency requirements specified in the IST Program. The current footnote (1) pertaining to the performance of recirculation spray pump 2RSS*P21A would be deleted.

Date of publication of individual notice in **Federal Register:** April 18, 1995 (60 FR 19417)

Expiration date of individual notice: May 18, 1995

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Maine Yankee Atomic Power Company, Docket No. 50-309, Maine Yankee Atomic Power Station, Lincoln County, Maine

Date of amendment request: April 14, 1995

Description of amendment request: The proposed amendment would revise the Technical Specifications to allow the use of the Westinghouse Electric Corporation sleeving process for repairing steam generator tubes.

Date of publication of individual notice in **Federal Register:** April 21, 1995 (60 FR 19969)

Expiration date of individual notice: May 22, 1995

Local Public Document Room location: Wiscasset Public Library, High Street, P.O. Box 367, Wiscasset, ME 04578.

Notice Of Issuance Of Amendments To Facility Operating Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License, Proposed No Significant Hazards Consideration Determination, and Opportunity for A Hearing in connection with these actions was published in the **Federal Register** as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved.

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: July 22, 1994, as supplemented on March 6, 1995

Brief description of amendments: The amendments change the Technical Specifications to implement a performance based assessment program, including corresponding organizational and functional changes. Specifically, the changes affect the independent review function, the independent assessment of plant activity and the Independent Safety Engineering Group. These functions will be performed by the Nuclear Assessment Section (NAS). The NAS's fundamental role will be to: (1) assist plant management in the early identification of issues that may prevent the plant from achieving quality, and (2) ensure effective correction of deficiencies.

Date of issuance: April 18, 1995

Effective date: April 18, 1995

Amendment Nos.: 177 and 208

Facility Operating License Nos. DPR-71 and DPR-62. Amendments revise the Technical Specifications.

Date of initial notice in **Federal Register:** August 31, 1994 (59 FR 45017) The March 6, 1995, submittal added Radiation Protection to the list of assessments in TS 6.5.5.2 and reworded Section 6.5.4.4, but did not change the no significant hazards consideration determination as published in the **Federal Register.** The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 18, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: June 18, 1992, as supplemented December 8, 1992 and February 3, 1995

Brief description of amendment: The amendment adds limiting conditions of operation and surveillance requirements for the pressurizer power-operated relief valves and their associated block valves whenever average temperature is above 350 degrees F or the reactor is critical. Specifications are also added for low-temperature overpressure protection whenever average temperature is less than 350 degrees F and the reactor coolant system is not vented to the containment.

Date of issuance: April 14, 1995

Effective date: April 14, 1995

Amendment No.: 162

Facility Operating License No. DPR-23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** September 2, 1992 (57 FR 40208). Renoticed on March 1, 1995 (60 FR 11127) The December 8, 1992, letter corrected a typographical error and did not affect the no significant hazards consideration. The licensee's letter dated February 3, 1995, proposed a revision to the TS regarding block valve testing in accordance with Generic Letter 90-06 recommendations. The proposed change was noticed on March 1, 1995 (60 FR 11127). The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 14, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: November 4, 1994, as supplemented April 6, 1995.

Brief description of amendment: The amendment changes the testing frequency of the turbine overspeed protection valves from monthly to quarterly to implement an enhancement recommended by Generic Letter 93-05, ''Line-Item Technical Specification Improvements to Reduce Surveillance Requirements for Testing During Power Operation.'' The April 6, 1995 submittal provided clarifying information only, and did not change the proposed no significant hazards determination.

Date of issuance: April 27, 1995

Effective date: April 27, 1995

Amendment No.: 164

Facility Operating License No. DPR-23. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** December 7, 1994 (59 FR 63115) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 27, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Carolina Power & Light Company, et al., Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: January 19, 1995, as supplemented March 20, 1995

Brief description of amendment: The amendment revises Technical Specification 4.0.3 and its associated Bases to provide for a delay period in which to perform a surveillance that was not performed within its specified frequency.

Date of issuance: April 17, 1995

Effective date: April 17, 1995

Amendment No.: 56

Facility Operating License No. NPF-63. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** February 15, 1995 (60 FR 8742) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 17, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Carolina Power & Light Company, et al., Docket No. 50-400, Shearon Harris Nuclear Power Plant, Unit 1, Wake and Chatham Counties, North Carolina

Date of application for amendment: July 22, 1994, as supplemented March 6, 1995.

Brief description of amendment: The amendment implements a performance-based assessment program, including corresponding organizational and functional changes. Specifically, the changes affect the Independent Review (IR) function, the independent assessment of plant activity and the Independent Safety Engineering Group. These functions will be performed by the proposed Nuclear Assessment Section (NAS). The NAS will perform internal evaluations and assessment activities and serve as plant management's staff for the objective oversight of plant performance relating to nuclear safety, reliability, and quality. The NAS's fundamental role will be to: (1) assist plant management in the early identification of issues which may prevent the plant from achieving quality performance on a sustained basis; and (2) ensure effective correction of deficiencies.

Date of issuance: April 21, 1995

Effective date: April 21, 1995

Amendment No.: 57

Facility Operating License No. NPF-63. Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** August 31, 1994 (59 FR 45019) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 21, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Cameron Village Regional Library, 1930 Clark Avenue, Raleigh, North Carolina 27605.

Commonwealth Edison Company, Docket No. 50-374, LaSalle County Station, Unit 2, LaSalle County, Illinois

Date of application for amendment: March 31, 1995

Brief description of amendment: The amendment revises the safety/relief valve (SRV) safety function lift setting allowable tolerance band from -3/+1% to plus or minus 3% and includes a requirement for the lift settings to be within plus or minus 1% of the technical specification limit following testing.

Date of issuance: April 25, 1995

Effective date: Immediately, to be implemented prior to restart from the sixth refueling outage.

Amendment No.: 89

Facility Operating License No. NPF-18: The amendment revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (60 FR 17590 dated April 6, 1995). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. This notice also provided for an opportunity to request a hearing by May 8, 1995, but indicated that if the Commission makes a final no significant hazards consideration determination, any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendment, finding of exigent circumstances, and final determination of no significant hazards consideration is contained in a Safety Evaluation dated April 25, 1995.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60690

Local Public Document Room location: Jacobs Memorial Library, Illinois Valley Community College, Oglesby, Illinois 61348.

Consolidated Edison Company of New York, Docket No. 50-247, Indian Point Nuclear Generating Unit No. 2, Westchester County, New York

Date of application for amendment: June 1, 1994, as supplemented on January 25, 1995, April 7, April 19, and April 26, 1995.

Brief description of amendment: The amendment revises Technical Specification Section 3.10 to allow extended Rod Position Indication (RPI) deviation limits and on-line calibration of the RPI channels for cycle 13 only.

Date of issuance: April 28, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 182

Facility Operating License No. DPR-26: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 20, 1994 (59 FR 37069). The January 25, April 7, April 19, and April 26, 1995, submittals provided clarifying information that did not affect the initial no significant hazards determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 28, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: White Plains Public Library, 100 Martine Avenue, White Plains, New York 10610.

Consumers Power Company, Docket No. 50-255, Palisades Plant, Van Buren County, Michigan

Date of application for amendment: February 10, 1995, as supplemented March 27 and 30, 1995

Brief description of amendment: This amendment revises the Technical Specifications to allow a one-time deferral of several 18-month interval surveillance tests until the upcoming scheduled refueling outage to avoid the necessity of imposing a plant shutdown solely for the sake of their performance. In the March 30, 1995, letter the license also withdrew its request for deferral of several surveillance tests.

Date of issuance: April 20, 1995

Effective date: April 20, 1995

Amendment No.: 164

Facility Operating License No. DPR-20. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11131) The March 27 and 30, 1995, letters provided clarifying information which was within the scope of the initial notice and did not affect the staff's original proposed no significant hazards consideration determination.The Commission's related evaluation of the amendment and of the withdrawalof certain surveillance test deferrals is contained in a Safety Evaluation dated April 20, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Van Wylen Library, Hope College, Holland, Michigan 49423.

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of application for amendments: February 23, 1995, as supplemented by letter dated March 21, 1995.

Brief description of amendments: The amendments revise Technical Specification (TS) 3.8.2.1 and TS 3.8.3.1 to allow installation of replacement equipment in response to an Electrical Distribution Systems Functional Inspection, conducted by the NRC in July 1991. The existing breaker arrangement could result in a trip of both the battery and main breakers if a fault occurs on one of the 125-V dc panelboards. The licensee committed to have these breakers replaced in 1995 with a better coordinated design to eliminate the concern.

Date of issuance: April 14, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment Nos.: 155 and 137

Facility Operating License Nos. NPF-9 and NPF-17: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 8, 1995 (60 FR 12791) The March 21, 1995, letter provided clarifying information that did not change the scope of the February 23, 1995, application and the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 14, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Duquesne Light Company, et al., Docket Nos. 50-334 and 50-412, Beaver Valley Power Station, Unit Nos. 1 and 2, Shippingport, Pennsylvania

Date of application for amendments: September 2, 1992

Brief description of amendments: These amendments revise the Appendix A Technical Specifications relating to the required surveillance frequency for comparing the incore and excore axial imbalance. The revision requires comparison of the incore to excore axial imbalance at least once every 31 Effective Full Power Days above 15 percent of rated thermal power rather than once every 31 days above 15 percent of rated thermal power as was previously required.

Date of issuance: April 26, 1995

Effective date: April 26, 1995

Amendment Nos.: 186 and 67

Facility Operating License Nos. DPR-66 and NPF-73: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 14, 1992 (57 FR 47128) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 26, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: B. F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Entergy Operations, Inc., Docket Nos. 50-313 and 50-368, Arkansas Nuclear One, Unit Nos. 1 and 2, Pope County, Arkansas

Date of amendment request: June 20, 1994

Brief description of amendments: The amendments relocated the requirements of the quality assurance program and the security and emergency plans from the administrative controls section of the technical specifications to the respective licensee-controlled documents.

Date of issuance: April 25, 1995

Effective date: 90 days from date of issuance

Amendment Nos.: 179 and 160

Facility Operating License Nos. DPR-51 and NPF-6. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 17, 1994 (59 FR 42340) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 25, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801.

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: August 5, 1993

Brief description of amendment: The amendment removed the requirements associated with loose-part detection system from the Technical Specifications for Waterford Steam Electric Station, Unit 3. These requirements will be incorporated into the Waterford 3 Updated Final Safety Analysis Report and maintained under the provisions of 10 CFR 50.59.

Date of issuance: April 20, 1995

Effective date: April 20, 1995

Amendment No.: 104

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 15, 1993 (58 FR 48382) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 20, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

Entergy Operations, Inc., Docket No. 50-382, Waterford Steam Electric Station, Unit 3, St. Charles Parish, Louisiana

Date of amendment request: April 4, 1995, as supplemented by letter dated April 5, 1995

Brief description of amendment: The amendment changed the Appendix A Technical Specifications (TSs) by revising the TSs for moderator temperature coefficient. The amendment approves a one time deviation by excluding the two-thirds end-of-cycle moderator temperature coefficient test requirement for Cycle 7.

Date of issuance: April 27, 1995

Effective date: April 27, 1995

Amendment No.: 105

Facility Operating License No. NPF-38. Amendment revised the Technical Specifications.Public comments requested as to proposed no significant hazards consideration: Yes (60 FR 18431, dated April 11, 1995). The notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination. No comments have been received. The notice also provided for an opportunity to request a hearing by May 11, 1995, but stated that any such hearing would take place after issuance of the amendment. The Commission's related evaluation of the amendments, finding of exigent circumstances, and final determination of no significant hazards consideration is contained in a Safety Evaluation dated April 27, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of New Orleans Library, Louisiana Collection, Lakefront, New Orleans, Louisiana 70122.

Entergy Operations, Inc., System Energy Resources, Inc., South Mississippi Electric Power Association, and Mississippi Power & Light Company, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: October 12, 1994

Brief description of amendment: The amendment removed License Condition 2.C.(26) related to Turbine Disk Integrity.

Date of issuance: April 17, 1995

Effective date: April 17, 1995

Amendment No: 121

Facility Operating License No. NPF-29. Amendment revises the license.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55868) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 17, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Judge George W. Armstrong Library, 220 S. Commerce Street, Natchez, MS 39120.

Florida Power and Light Company, et al., Docket Nos. 50-335 and 50-389, St. Lucie Plant, Unit Nos. 1 and 2, St. Lucie County, Florida

Date of application for amendments: May 23, 1994

Brief description of amendments: These amendments will relocate the seismic monitoring instrumentation Limiting Conditions of Operation, Surveillance Requirements and the associated tables contained in Technical Specifications 3.3.3.3, 4.3.3.3.1 and 4.3.3.3.2 to the Updated Final Analysis Report.

Date of issuance: April 25, 1995

Effective date: April 25, 1995

Amendment Nos.: 135 and 74

Facility Operating License Nos. DPR-67 and NPF-16: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** July 6, 1994 (59 FR 34664) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 25, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Fort Pierce, Florida 34954-9003.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50-424 and 50-425, Vogtle Electric Generating Plant, Units 1 and 2, Burke County, Georgia

Date of application for amendments: January 20, 1995

Brief description of amendments: The amendments revise the administrative requirements of Technical Specification (TS) 6.4.1.2 related to the areas of technical expertise that must be represented on the Plant Review Board (PRB). The licensee proposed this change in order to maintain an appropriate level of PRB expertise after the implementation of a planned reorganization that includes combining certain departments that are listed separately in the current TS 6.4.2.1 requirements.

Date of issuance: April 27, 1995

Effective date: As of the date of issuance to be implemented within 30 days

Amendment Nos.: 84 and 62

Facility Operating License Nos. NPF-68 and NPF-81: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 6, 1995 (60 FR 7077) The April 4, 1995, letter provided additional and clarifying information that did not change the scope of the January 20, 1995, application or the initial proposed no significant hazards consideration determination.The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 27, 1995. No significant hazards consideration comments received: No

Local Public Document Room location: Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Houston Lighting & Power Company, City Public Service Board of San Antonio, Central Power and Light Company, City of Austin, Texas, Docket Nos. 50-498 and 50-499, South Texas Project, Units 1 and 2, Matagorda County, Texas

Date of amendment request: November 8, 1994, as supplemented by letter dated March 14, 1995.

Brief description of amendments: The amendments require that only one of the two battery chargers associated with each Class 1E 125-VDC Channel I and Channel IV is operable.

Date of issuance: April 17, 1995

Effective date: April 17, 1995, to be implemented within 31 days.

Amendment Nos.: Unit 1 - Amendment No. 73; Unit 2 - Amendment No. 62

Facility Operating License Nos. NPF-76 and NPF-80. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 7, 1994 (59 FR 63123) The March 14, 1995, supplement withdrew that portion of the proposed amendments where the required wording was already incorporated into the Technical Specifications by amendments issued on February 14, 1995, in response to another amendment request. The March 14, 1995, letter also provided clarifying information and did not change the original no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 17, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Wharton County Junior College, J. M. Hodges Learning Center, 911 Boling Highway, Wharton, TX 77488.

IES Utilities Inc., Docket No. 50-331, Duane Arnold Energy Center, Linn County, Iowa

Date of application for amendment: November 10, 1994, as supplemented March 1, 1995

Brief description of amendment: The amendment revises the Duane Arnold Energy Center Technical Specification Section 3.2.A to refer to the Offsite Dose Assessment Manual for the setpoint of the Offgas Stack Radiation Monitor and makes the ''Applicable Operating Mode'' and the ''Action'' statements for these instruments consistent with the required function. The Action statement for the other instruments which initiate Secondary Containment isolation is also revised to be consistent with the current practice and with the function of those instruments. The Basis is also revised to add further description of the function and requirements.

Date of issuance: April 25, 1995

Effective date: April 25, 1995

Amendment No.: 209

Facility Operating License No. DPR-49. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 21, 1994 (59 FR 65815) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 25, 1995.The March 1, 1995, submittal provided supplemental information that did not change the initial proposed no significant hazards consideration determination.No significant hazards consideration comments received: No.

Local Public Document Room location: Cedar Rapids Public Library, 500 First Street, S.E., Cedar Rapids, Iowa 52401.

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: April 6, 1994

Brief description of amendments: The amendments delete part of License Condition 2.C.(4) to Operating License No. DPR-58 and part of License Condition 2.C.(3)(o) to Operating License No. DPR-74 on fire protection. The related fire protection safety evaluation also changes three of the modifications listed in Table 1 of the Safety Evaluation Report of July 31, 1979, that supported amendments nos. 31 and 12 to Operating Licenses No. DPR-58 and No. DPR-74, respectively.

Date of issuance: April 19, 1995

Effective date: April 19, 1995

Amendment Nos.: 194 and 180

Facility Operating License Nos. DPR-58 and DPR-74. Amendments revised the Facility Operating Licenses.

Date of initial notice in **Federal Register:** September 28, 1994 (59 FR 49429) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 19, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Maud Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit 2, Oswego County, New York

Date of application for amendment: March 9, 1995

Brief description of amendment: The amendment revises Technical Specification Section 4.6.1.2.a, Primary Containment/Containment Leakage. This change allows the second Type A containment leak rate test to be performed at refueling outage 5 instead of refueling outage 4, consistent with an exemption to 10 CFR Part 50, Appendix J which has been granted.

Date of issuance: April 24, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 65

Facility Operating License No. NPF-69: Amendment revises the Technical Specifications.

Date of initial notice in **Federal Register:** March 23, 1995 (60 FR 15310) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: December 2, 1994

Brief description of amendment: The amendment changes the Millstone 3 Technical Specification Table 4.3-1 by adding a note for certain Functional Units which would allow an entry into Mode 2 or Mode 1 before performing calibration for the power range detectors.

Date of issuance: April 26, 1995

Effective date: As of the date of issuance to be implemented within 30 days.

Amendment No.: 109

Facility Operating License No. NPF-49. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** February 1, 1995 (60 FR 6304) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 26, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, Thames Valley Campus, 574 New LondonTurnpike, Norwich, CT 06360.

Northern States Power Company, Docket Nos. 50-282 and 50-306, Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, Goodhue County, Minnesota

Date of application for amendments: February 23 and March 3, 1995

Brief description of amendments: The amendments revise the Prairie Island Technical Specifications section 4.4.A.5 to add the phrase ''and all approved exemptions.'' after the reference to 10 CFR Part 50, Appendix J. This revision will allow implementation of approved exemptions from the testing schedule requirements of 10 CFR Part 50, Appendix J, Section III.D.1.(a).

Date of issuance: April 18, 1995

Effective date: April 18, 1995, with full implementation within 30 days.

Amendment Nos.: 117 and 110

Facility Operating License Nos. DPR-42 and DPR-60. Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 15, 1995 (60 FR 14025). The March 3, 1995, letter provided clarifying information within the scope of the original submittal and did not change the staff's initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 18, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Minneapolis Public Library, Technology and Science Department, 300 Nicollet Mall, Minneapolis, Minnesota 55401.

Pacific Gas and Electric Company, Docket Nos. 50-275 and 50-323, Diablo Canyon Nuclear Power Plant, Unit Nos. 1 and 2, San Luis Obispo County, California

Date of application for amendments: August 17, 1994 (Reference LAR 94-06)

Brief description of amendments: The proposed amendments increase the allowed outage time of the refueling water storage tank (RWST) for adjustment of boron concentration from one to eight hours as contained in Technical Specifications Section 3.5.5.

Date of issuance: April 14, 1995

Effective date: April 14, 1995, to be implemented within 30 days of issuance

Amendment Nos.: Unit 1 - Amendment No. 101; Unit 2 - Amendment No. 100

Facility Operating License Nos. DPR-80 and DPR-82: The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 12, 1994 (59 FR 51621) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 14, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: California Polytechnic State University, Robert E. Kennedy Library, Government Documents and Maps Department, San Luis Obispo, California 93407.

PECO Energy Company, Public Service Electric and Gas Company Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: October 25, 1994 as supplemented February 13, 1995.

Brief description of amendments: The amendment clarifies the technical specification surveillance requirements and bases for high pressure coolant injection system testing at low reactor pressure.

Date of issuance: April 18, 1995

Effective date: April 18, 1995Amendments Nos.: 200 and 202

Facility Operating License Nos. DPR-44 and DPR-56: The amendments revised the Technical Specifications. Public comments requested as to proposed no significant hazards consideration: Yes (59 FR 55498 dated November 7, 1994). That notice provided an opportunity to submit comments on the Commission's proposed no significant hazards consideration determination, and also provided an opportunity to request a hearing by December 7, 1994. No comments or requests for hearings have been received. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 18, 1995.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket No. 50-278, Peach Bottom Atomic Power Station, Unit No. 3, York County, Pennsylvania

Date of application for amendment: January 13, 1995 as supplemented by letters dated March 14, 1995 and April 12, 1995.

Brief description of amendment: The requested changes would modify Tables 3.7.1 and 3.7.4 of the Technical Specifications (TS) to reflect a change in the number of primary containment penetrations and isolation valves associated with the traversing in-core probe (TIP) system. In order to prevent confusion with the staff's review of PECO's September 29, 1994 application to implement improved TS at Peach Bottom, the staff is issuing the license amendment regarding the TIP system for Unit 3 only.

Date of issuance: April 24, 1995

Effective date: April 24, 1995

Amendment No.: 203

Facility Operating License No. DPR-56: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 1, 1995 (60 FR 11139) The March 14, 1995 and April 12, 1995, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 24, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, (REGIONAL DEPOSITORY) Education Building, Walnut Street and Commonwealth Avenue, Box 1601, Harrisburg, Pennsylvania 17105.

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: September 20, 1994

Brief description of amendments: The amendments modify the Technical Specifications for auxiliary feedwater to reduce the secondary side steam pressure required for testing the turbine driven auxiliary feedwater pump and to allow 24 hours to perform the test after reaching the minimum test pressure.

Date of issuance: April 17, 1995

Effective date: As of the date of issuance to be implemented within 60 days.

Amendment Nos.: 165 and 146

Facility Operating License Nos. DPR-70 and DPR-75. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** November 9, 1994 (59 FR 55889) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 17, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

Public Service Electric & Gas Company, Docket No. 50-311, Salem Nuclear Generating Station, Unit No. 2, Salem County, New Jersey

Date of application for amendment: February 3, 1994, as supplemented September 19, 1994, and November 23, 1994

Brief description of amendment: The amendment changes the Technical Specifications to reflect a reduction in Reactor Coolant System flow.

Date of issuance: April 17, 1995

Effective date: April 17, 1995

Amendment No.: 147

Facility Operating License No. DPR-75: This amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 15, 1995 (60 FR 14028) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 17, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

Rochester Gas and Electric Corporation, Docket No. 50-244, R. E. Ginna Nuclear Power Plant, Wayne County, New York

Date of application for amendment: March 13, 1995

Brief description of amendment: This amendment revises Technical Specification 4.4.2.4.a to replace specific leakage rate testing frequencies for containment isolation valves that require Type C testing for the 1995 refueling outage to be completed prior to exiting Cold Shutdown tentatively scheduled for April 27, 1995.

Date of issuance: April 26, 1995

Effective date: April 26, 1995

Amendment No.: 59

Facility Operating License No. DPR-18: Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 22, 1995 (60 FR 15167) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 26, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Rochester Public Library, 115 South Avenue, Rochester, New York 14610.

TU Electric Company, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: February 23, 1994 (LAR 94-005, TXX-94034)

Brief description of amendments: These amendments changed Technical Specification (TS) 3/4.5.1, ''Emergency Core Cooling Systems, Accumulators, Cold Leg Injection,'' to: 1) allow a one hour allowed outage time following discovery of a closed cold leg injection accumulator discharge isolation valve in Modes 1, 2, or 3; 2) eliminate the redundant requirement to reverify accumulator boron concentration following fill from the refueling water storage tank RWST; 3) remove the accumulator water level and pressure channel analog channel operational test and channel calibration from the TSs; and 4) change the accumulator limits to analysis values rather than indicated values. Also these amendments modified TS 3/4.5.2, ''ECCS Subsystems - Tavg ≤ 350°F'' to reduce the visual inspection frequency following containment entries.

Date of issuance: April 27, 1995

Effective date: April 27, 1995, to be implemented within 30 days.

Amendment Nos.: Unit 1 - Amendment No. 40; Unit 2 - Amendment No. 26

Facility Operating License Nos. NPF-87 and NPF-89. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** August 3, 1994 (59 FR 39597) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 27, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

TU Electric Company, Docket Nos. 50-445 and 50-446, Comanche Peak Steam Electric Station, Unit Nos. 1 and 2, Somervell County, Texas

Date of amendment request: August 9, 1994, (LAR 94-013, TXX-94211)

Brief description of amendments: These amendments eliminated ''High Negative Neutron Flux Rate'' reactor trip function based on analyses which demonstrate that the protection provided by the reactor trip function is not required. The affected Technical Specifications were: 2.2.1, ''Reactor Trip System Instrumentation Setpoints,'' and 3/4.3.1, ''Reactor Trip System Instrumentation.'' Also affected was Bases Section 2.2.1.

Date of issuance: April 17, 1995

Effective date: April 17, 1995, to be implemented within 30 days.

Amendment Nos.: Unit 1 - Amendment No. 39; Unit 2 - Amendment No. 25

Facility Operating License Nos. NPF-87 and NPF-89. The amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** September 28, 1994 (59 FR 49438) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated April 17, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of Texas at Arlington Library, Government Publications/Maps, 702 College, P.O. Box 19497, Arlington, TX 76019.

Union Electric Company, Docket No. 50-483, Callaway Plant, Unit 1, Callaway County, Missouri

Date of application for amendment: September 9, 1994, as supplemented on December 22, 1994

Brief description of amendment: The amendment revises the Technical Specification (TS) 3/4.8.2.1, 3/4.8.2.2, 3/4.8.3.1, and 3/4.8.3.2. The changes address the 125-volt DC buses and adds provisions for swing battery chargers, and removes provisions for the 4160-volt and 480-volt AC emergency buses.

Date of issuance: April 18, 1995

Effective date: April 18, 1995

Amendment No.: 99

Facility Operating License No. NPF-30. Amendment revises the Technical Specification Bases and FSAR.

Date of initial notice in **Federal Register:** January 4, 1995 (60 FR 506) The December 22, 1994, letter provided supplemental information that did not change the initial proposed no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 18, 1995. No significant hazards consideration comments received: No.

Local Public Document Room location: Callaway County Public Library, 710 Court Street, Fulton, Missouri 65251.

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of application for amendments: February 14, 1995

Brief description of amendments: These amendments modify the Technical Specifications (TS) to revise Section 4.4.D of the TS to permit approved exemptions to the containment integrated leak rate test frequency requirements.

Date of issuance: April 18, 1995

Effective date: April 18, 1995

Amendment Nos.: 196 and 196

Facility Operating License Nos. DPR-32 and DPR-37: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** March 15, 1995 (60 FR 14029) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 18, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Unit Nos. 1 and 2, Surry County, Virginia.

Date of application for amendments: September 6, 1994, as supplemented March 7, 1995

Brief description of amendments: These amendments modify the Technical Specifications to revise the review responsibilities of the Station Nuclear Safety and Operating Committee and the Management Safety Review Committee.

Date of issuance: April 21, 1995

Effective date: April 21, 1995

Amendment Nos.: 197 and 197

Facility Operating License Nos. DPR-32 and DPR-37: Amendments revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 12, 1994 (59 FR 51631) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 21, 1995.No significant hazards consideration comments received: No

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Washington Public Power Supply System, Docket No. 50-397, Nuclear Project No. 2, Benton County, Washington

Date of application for amendment: April 1, 1993

Brief description of amendment: This amendment revises TS 3.8.1, ''A.C. Sources'' by increasing the minimum required level of diesel generator fuel storage capacity. This change is based on testing and revised calculations that demonstrated that the existing levels of DG fuel storage were inadequate to meet the post-loss of coolant accident fuel consumption requirements for seven days of operation.

Date of issuance: April 25, 1995

Effective date: April 25, 1995, to be implemented within 30 days of issuance

Amendment No.: 136

Facility Operating License No. NPF-21: The amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** May 12, 1993 (58 FR 28065) The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 25, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: Richland Public Library, 955 Northgate Street, Richland, Washington 99352.

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee NuclearPower Plant, Kewaunee County, Wisconsin

Date of application for amendment: August 24, 1994 as supplemented on January 23, 1995.

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 3.1.b.1 and Figure TS 3.1-4 regarding Low Temperature Overpressure (LTOP) protection for the reactor coolant pressure boundary. The change extends the LTOP requirements through the end of operating cycle 21 or 18.40 effective full power years. The Basis Section has also been modified to reflect these changes.

Date of issuance: April 26, 1995

Effective date: April 26, 1995

Amendment No.: 120

Facility Operating License No. DPR-43. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 12, 1994 (59 FR 51632). The January 23, 1995, submittal, provided additional reference material which did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 26, 1995.No significant hazards consideration comments received: None.

Local Public Document Room location: University of Wisconsin Library Learning Center, 2420 Nicolet Drive, Green Bay, Wisconsin 54301.

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: November 8, 1994, as supplemented on January 9, February 14, March 8, and April 3, 1995.

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 3.1.d, ''Leakage of Reactor Coolant,'' TS 4.2.b, ''Steam Generator Tubes,'' and TS 3.4.a, ''Steam Generators,'' to allow application of a voltage-based repair limit for the steam generator (SG) tube support plate (TSP) intersections experiencing outside diameter stress corrosion cracking (ODSCC). The amendment also reduces the allowed primary-to-secondary operational leakage from any one SG from 500 gallons per day (gpd) to 150 gpd. These changes to the tube repair criteria are applicable for the 1995 to 1996 operating cycle (Cycle 21) only.

Date of issuance: April 17, 1995

Effective date: April 17, 1995

Amendment No.: 118

Facility Operating License No. DPR-43. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** December 7, 1994 (59 FR 63127). The January 9, February 14, and March 8, and April 3, 1995, submittals provided clarifying information which did not change the initial no significant hazards consideration determination. The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 17, 1995. No significant hazards consideration comments received: None.

Local Public Document Room location: University of Wisconsin Library Learning Center, 2420 Nicolet Drive, Green Bay, Wisconsin 54301.

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: September 7, 1994

Brief description of amendment: The amendment revises Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS) by adding two new sections, TS Section 3.0 and TS Section 4.0, with associated bases. TS Section 3.0 establishes the general requirements applicable to each of the Limiting Conditions for Operation (LCOs) within Section 3 of the KNPP TS. TS Section 4.0 establishes the general requirements applicable to Surveillance Requirements. The new requirements of TS 4.0.b also affect TS Sections 4.5, 4.6, 4.7, and Tables TS 4.1-2 and 4.1-3.

Date of issuance: April 18, 1995

Effective date: April 18, 1995

Amendment No.: 119

Facility Operating License No. DPR-43. Amendment revised the Technical Specifications.

Date of initial notice in **Federal Register:** October 12, 1994 (59 FR 51632)The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated April 18, 1995.No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin Library Learning Center, 2420 Nicolet Drive, Green Bay, Wisconsin 54301.

Notice Of Issuance Of Amendments To Facility Operating Licenses And Final Determination Of No Significant Hazards Consideration And Opportunity For A Hearing (Exigent Public Announcement Or Emergency Circumstances)

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Because of exigent or emergency circumstances associated with the date the amendment was needed, there was not time for the Commission to publish, for public comment before issuance, its usual 30-day Notice of Consideration of Issuance of Amendment, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing.

For exigent circumstances, the Commission has either issued a **Federal Register** notice providing opportunity for public comment or has used local media to provide notice to the public in the area surrounding a licensee's facility of the licensee's application and of the Commission's proposed determination of no significant hazards consideration. The Commission has provided a reasonable opportunity for the public to comment, using its best efforts to make available to the public means of communication for the public to respond quickly, and in the case of telephone comments, the comments have been recorded or transcribed as appropriate and the licensee has been informed of the public comments.

In circumstances where failure to act in a timely way would have resulted, for example, in derating or shutdown of a nuclear power plant or in prevention of either resumption of operation or of increase in power output up to the plant's licensed power level, the Commission may not have had an opportunity to provide for public comment on its no significant hazards consideration determination. In such case, the license amendment has been issued without opportunity for comment. If there has been some time for public comment but less than 30 days, the Commission may provide an opportunity for public comment. If comments have been requested, it is so stated. In either event, the State has been consulted by telephone whenever possible.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendment involves no significant hazards consideration. The basis for this determination is contained in the documents related to this action. Accordingly, the amendments have been issued and made effective as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the application for amendment, (2) the amendment to Facility Operating License, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment, as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room for the particular facility involved.

The Commission is also offering an opportunity for a hearing with respect to the issuance of the amendment. By June 9, 1995, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written request for a hearing and a petition for leave to intervene. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's ''Rules of Practice for Domestic Licensing Proceedings'' in 10 CFR Part 2. Interested persons should consult a current copy of 10 CFR 2.714 which is available at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC and at the local public document room for the particular facility involved. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) the nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to 15 days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than 15 days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter. Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the petitioner shall provide a brief explanation of the bases of the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the petitioner intends to rely in proving the contention at the hearing. The petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the petitioner intends to rely to establish those facts or expert opinion. Petitioner must provide sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the petitioner to relief. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses. Since the Commission has made a final determination that the amendment involves no significant hazards consideration, if a hearing is requested, it will not stay the effectiveness of the amendment. Any hearing held would take place while the amendment is in effect.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date. Where petitions are filed during the last 10 days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 248-5100 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number N1023 and the following message addressed to **(Project Director)**: petitioner's name and telephone number, date petition was mailed, plant name, and publication date and page number of this **Federal Register** notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for a hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board that the petition and/or request should be granted based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

Carolina Power & Light Company, Docket No. 50-261, H. B. Robinson SteamElectric Plant, Unit No. 2, Darlington County, South Carolina

Date of application for amendment: April 13, 1995, as supplemented April 18, 1995.

Brief description of amendment: Amendment revises TS Section 4.4.3.f, g, and h to allow the post accident heat removal system surveillance test interval to be changed from a 12-month interval to a refueling outage interval.

Date of issuance: April 19, 1995

Effective date: April 19, 1995

Amendment No.: 163

Facility Operating License No. DPR-23. Amendment revises the Technical Specifications.The Commission's final determination of significant hazards consideration and related evaluation of the amendment is contained in a Safety Evaluation dated April 19, 1995.

Local Public Document Room location: Hartsville Memorial Library, 147 West College Avenue, Hartsville, South Carolina 29550.

Dated at Rockville, Maryland, this 3rd day of May, 1995.

For The Nuclear Regulatory Commission

Elinor G. Adensam,

Acting Director, Division of Reactor Projects - III/IV, Office of Nuclear Reactor Regulation

[Doc. 95–11367 Filed 5–9–95; 8:45 am]

BILLING CODE 7590–01–F

[Docket No. 50–160–Ren; ASLBP No. 95–704–01–Ren]

Georgia Institute of Technology, Atlanta, GA, Georgia Tech Research Reactor, (Renewal of Facility License No. R–97); Notice of Hearing

May 4, 1995.

On September 26, 1994, the Nuclear Regulatory Commission published in the **Federal Register** a notice of opportunity for hearing with respect to the proposed renewal of the facility operating license for the Georgia Tech Research Reactor, located on the campus of the Georgia Institute of Technology in Atlanta, Georgia (59 FR 49088. One request for a hearing and petition for leave to intervene, filed by Georgians Against Nuclear Energy (GANE), was received. On November 18, 1994, an Atomic Safety and Licensing Board was established to rule upon this request and to preside over the proceeding in the event that a hearing were ordered.

After holding a prehearing conference in Atlanta, Georgia, the Atomic Safety and Licensing Board issued a Prehearing Conference Order (LBP–95–6) on April 26, 1995, granting GANE's request for a hearing and petition for leave to intervene.

Please take notice that a hearing will be conducted it this proceeding. The Atomic Safety and Licensing Board designated to preside over the proceeding consists of Dr. Jerry R. Kline, Dr. Peter S. Lam, and Charles Bechhoefer, who will serve as Chairman of the Board.

During the course of the proceeding, the Board may hold one or more prehearing conferences pursuant to 10 CFR 2.752 and, if necessary, an evidentiary hearing. The public is invited to attend all these sessions, except to the extent that information protected by 10 CFR 2.790 (relevant to one of the contentions accepted by the Board) may be discussed.

Supplementing the opportunity afforded at the first prehearing conference, during some or all of these sessions, and in accordance with 10 CFR 2.715(a), any person not a party to the proceeding will be permitted to make a limited appearance statement, either in writing or (depending on time availability) orally, setting forth his or her position on the issues. These statements do not constitute testimony or evidence in these proceedings but may assist the Board and/or parties in the definition of issues being considered. To the extent that oral statements are permitted, the number of persons making such statements and the time allotted for each may be limited depending upon the time available at various sessions. Written statements may be submitted at any time. Written statements, and requests to make oral limited appearance statements, should be submitted to the Secretary, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Docketing and Service Branch. A copy of such statement or request should be served on the Chairman of this Atomic Safety and Licensing Board, T3 F23, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

Documents relating to this proceeding are available for public inspection at the Commission's Public Document Room, 2120 L St. N.W., Washington, D.C. 20555.

Rockville, MD, May 4, 1995.

For the Atomic Safety and Licensing Board.

Charles Bechhoefer,

Chairman, Administrative Judge.

[FR Doc. 95–11532 Filed 5–9–95; 8:45 am]

BILLING CODE 7590–01–M

[Docket Nos. 70–7001; 70–7002]

United States Enrichment Corporation: Paducah Gaseous Diffusion Plant; Portsmouth Gaseous Diffusion Plant; Notice of Cancellation of Comment Period and Cancellation of Public Meetings Due to Inadequate Application for Certification

The U.S. Nuclear Regulatory Commission (NRC) received by letter dated April 18, 1995, an application from the United States Enrichment Corporation (USEC) for the initial certification of the gaseous diffusion plants (GDPs) located near Paducah, Kentucky and Piketon, Ohio. Notice of receipt of this application along with notice of comment period and public meetings was published in The **Federal Register** on April 28, 1995 (60 FR 21011). However, NRC's preliminary